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FINAL ENVIRONMENTAL IMPACT STATEMENT

on a

Proposed Nuclear Weapons Nonproliferation Policy Concerning Foreign Research Reactor Spent Nuclear Fuel

Appendix F
Description and Impacts of Storage Technology Alternatives

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Appendix F
Description and Impacts of Storage Technology Alternatives

Summary

This appendix presents a description and evaluation of currently available spent nuclear fuel storage technologies, and their applicability to foreign research reactor spent nuclear fuel. These technologies represent the range of alternatives that would be available to implement the proposed action. Some of these technologies are currently in use at U.S. Department of Energy (DOE) facilities. Several dry storage cask and/or building designs have been licensed by the U.S. Nuclear Regulatory Commission (NRC) and are operational with commercial nuclear power plant spent fuel at several locations.

This appendix also discusses potential storage sites and impacts of foreign research reactor spent nuclear fuel storage at these locations. The major sections in this appendix are:

Section F.1 Description of Existing and Proposed Technologies for Storage of Spent Nuclear Fuel

Section F.2 Storage Technology Evaluation Methodology

Section F.3 Selection of Storage Technologies for Further Evaluation

Section F.4 Environmental Impacts at Foreign Research Reactor Spent Nuclear Fuel Management Sites

Section F.5 Occupational Radiation Impacts from Receipt and Handling of Foreign Research Reactor Spent Nuclear Fuel

Section F.6 Evaluation Methodologies and Assumptions for Incident-Free Operations and Hypothetical Accidents at Management Sites

Section F.7 Economic Evaluation of Foreign Research Reactor Spent Nuclear Fuel Storage and Related Management Alternatives

Figure F-1 presents the different spent nuclear fuel storage technologies, which are divided into wet and dry systems and further classified by their materials of construction (i.e., concrete, metal), location (i.e., aboveground or belowground), and size (i.e., cask versus vault building or pool). The final level of detail is the specific design with 12 specific vendors' designs displayed in this figure. The following specific designs are of U.S. origin: Nuclear Assurance Corporation, MC-10, NUHOMS, and Ventilated Storage Cask-24 (Section F.1 describes these in more detail). The others are designed by foreign companies, but many of these companies, such as Transnuclear Inc., have U.S. affiliates. The principal categories of spent nuclear fuel storage technology are dry vault (building), dry cask, and wet pool.

This appendix discusses the aforementioned designs in terms of their shielding, criticality, thermal, structural, cost, and ease of use features. Some numerical design parameters are presented for comparison. Advantages and vulnerabilities of each design are also presented. Since none of these designs have been specifically designed or licensed for foreign research reactor spent nuclear fuel and related research reactor type fuel, some extrapolation has been made in this comparative assessment. All of the existing commercial designs and proposed new designs appear to be adaptable to foreign research reactor spent

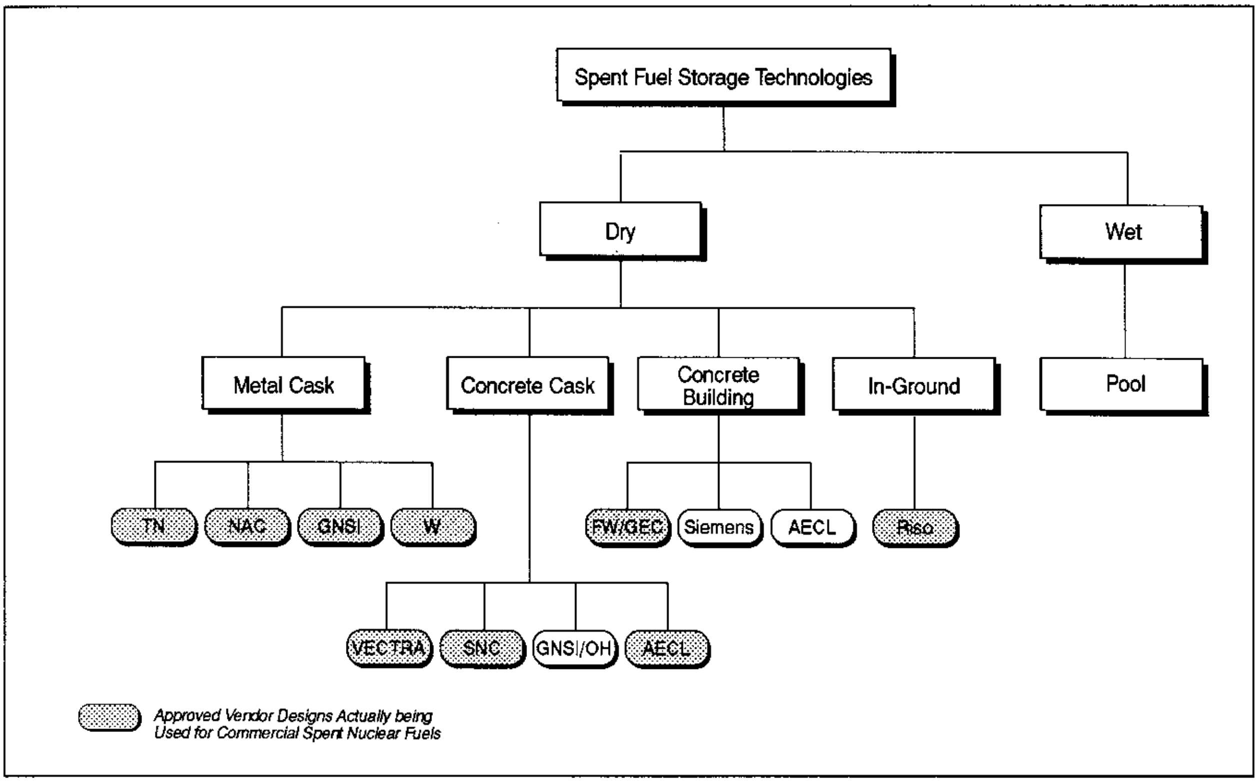


Figure F-1 Spent Nuclear Fuel Storage Technologies and Vendors

nuclear fuel with only minor, easily implemented modifications, such as interior baskets for holding the spent nuclear fuel. Use of existing facilities at a site for staging and characterization favors a cask storage approach, while a stand-alone, separate spent nuclear fuel storage approach requires a vault and other support facilities. Schedule and monetary considerations favor casks over the vault for sites with existing facilities, and this is why most domestic utilities are pursuing dry casks for long-term storage of spent nuclear fuel. Casks are the only independent spent nuclear fuel storage installation designs that have received certification by the NRC in accordance with 10 Code of Federal Regulations (CFR) 72 Appendix K.

The evaluation indicates that both wet and dry storage of foreign research reactor spent nuclear fuel appear acceptable for the time periods envisioned for the proposed action (i.e., through 2036). Commercial spent nuclear fuel dry storage systems require a minimum wet pool storage time or cooldown period of approximately 5 years after discharge from the nuclear reactor prior to emplacement into dry storage. In actual practice, this usually averages around an 8-year average cooldown period and, frequently, the commercial spent nuclear fuel has had over a 10-year cooldown period in a wet pool prior to emplacement into dry storage. This cooldown period ensures that licensed conditions for cladding temperatures (based upon potential corrosion, and usually around 350°C, or 630°F) are not exceeded. Foreign research reactor spent nuclear fuel has a lower cladding temperature limit based upon a phase transition in the aluminum metal cladding; this aluminum cladding limit has been identified as 175°C (315°F). Thus, a maximum cooldown period of 3 years of wet pool storage after irradiation has been identified for the foreign research reactor spent nuclear fuel prior to emplacement into dry storage. This would ensure that all foreign research reactor spent nuclear fuel elements were below a heat load of 40 Watts each, and most elements would be 10 Watts or less. The majority of the currently available foreign research reactor spent nuclear fuel already meets this requirement. Most of the existing dry storage designs appear acceptable for foreign research reactor spent nuclear fuel, without any clear preference. It should be noted that a research and development project to examine the applicability of aluminum-clad spent nuclear fuel dry storage at the Savannah River Site was initiated in Fiscal Year (FY) 1994.

The utilization of dry storage methods for foreign research reactor spent nuclear fuel requires the acquisition of racks, baskets, storage canisters, and/or casks. New construction would be required for dry vaults, except for several existing facilities at the Nevada Test Site and Idaho National Engineering Laboratory.

The utilization of wet storage methods requires a lined basin within a seismically qualified facility with the ability to maintain water chemistry and handle liquid radioactive waste. Currently, there are few existing DOE facilities in this category, and none have sufficient capacity to accommodate all of the foreign research reactor spent nuclear fuel. Thus, the selection of wet storage would require DOE acquisition of a facility, either by new construction or purchase of an existing facility such as the Barnwell Nuclear Fuels Plant (BNFP) that is owned by Allied General Nuclear Services. A summary of storage technology characteristics is given Table F-1. Sections F.4, F.5, and F.6 address environmental impacts, occupational dose, and accident consequences for storage. Section F.7 discusses costs in detail.

Table F-1 Summary of Storage Technology Characteristics for Commercial and Foreign Research Reactor Spent Nuclear Fuel

Storage Technology	*DOE Site Status (New or Existing)*	*Land Use Ha (Ac)*	*Annual Low-level Waste (m^3)*[a]	*Potential Annual Spent Nuclear Fuel Storage Public Impact (LCFs)*	*Lead Time until Spent Nuclear Fuel Storage (Years)*
Dry Vault - Utility Fuel	New	4 (5)	1-4	NA	2-3
Dry Cask (Concrete) - Utility Fuel	New	4 (5)	1-4	NA	2-3
Wet Pool - Utility Fuel	New	2 (3)	1-4	NA	3-5
Dry Vault - Savannah River Site Research Reactor Fuel	New	4 (8)	16	0	5-10[b]
Dry Cask - Savannah River Site Research Reactor Fuel	New	4 (5)	16	0	3-5[b]
Wet Pool - Idaho National Engineering Laboratory Research Reactor Fuel	New	2 (3)	12	2.4×10^{-12} to 2.5×10^{-10}	5-10[b]

NA = Not Available; LCF = Latent Cancer Fatality

[a] *Low-Level Waste generation decreases significantly if spent nuclear fuel is only being stored, without additional spent nuclear fuel receipts. To convert to ft^3, multiply by 35.3.*

[b] *To allow for extended periodic examination and characterization of fuel.*

DOE currently has pilot-scale experience with dry storage of spent nuclear fuel, and there are no identified technical constraints that would prevent dry storage of foreign research reactor spent nuclear fuel. There would be some need, however, for characterization, canning, and periodic inspection and monitoring. Both NRC-licensed and not yet licensed dry storage designs are readily available from commercial vendors. NRC-licensed designs have the following advantages:

- specific NRC requirements have been met that are equivalent to DOE requirements and guidance,
- extensive, interactive technical safety reviews have already been conducted between the supplier and the regulator,
- peer and public review has occurred as part of the licensing process,
- proven applications are in operation at commercial nuclear power plant sites, and
- documentation and quality assurance requirements have been satisfied.

For sites with an existing spent nuclear fuel infrastructure that includes facilities for spent nuclear fuel receipt, examination, and loading, a modular approach based upon casks can be implemented rapidly to meet Phase 1 requirements using standard funding and procurement capital appropriation methods. The casks could also be used for Phase 2, and their usage would avoid additional procurement. A modular dry vault approach represents an integrated self-contained, stand-alone facility, and can be used at any of the proposed management sites. However, construction of the vault could represent a major project or major systems acquisition under DOE management requirements, which may require a 7 to 10 year period for completion. Thus, a vault dry storage approach probably could not be available immediately. Metal cask development programs, such as dual- and multi-purpose casks, eliminate many storage site handling requirements and may provide future improvements.

Section F.7 evaluates the economics of the entire (40-year plus) foreign research reactor spent nuclear fuel program, including transportation, receipt/handling/inspection, storage, preparation for disposal, transportation to the repository, and disposal for the storage/disposal and chemical separation/vitrification alternatives. Costs are presented as rough-order-of-magnitude net present values, using a 4.9 percent real discount rate. In 1996 dollars, minimum total program costs for the storage alternative are about $800 million. This total divides into four very roughly equal parts: shipping to the United States and program management; receiving and storage at existing facilities; receiving, storage, and fuels qualification at not-yet existing facilities; and repository disposal. Other cost factors would be expected to add as much as $500 million to the program costs. Among the other cost factors are systems integration and logistics contingencies ($75 to $100 million), risks associated with limited characterization of the spent nuclear fuel ($100 million), risks associated with direct disposal of HEU ($50 to $100 million) and the probability that future discount rates will be lower than the current 4.9 percent rate ($200 million or more). Total costs, including all contingencies and risks could thus be in the $1.3 billion range.

For chemical separation alternatives, minimum total program costs are about $700 million. Savings in chemical separation and disposal of high-level waste versus storage and disposal of spent nuclear fuel account for the bulk of the difference between the costs in the chemical separation case and the storage case. Other cost factors would be expected to add as much as $250 million to the program costs. The key cost factors are systems integration and logistics contingencies ($75 to $100 million) and the probability that future discount rates will be lower than the current 4.9 percent rate ($100 million or more). If part of the material shipped to the Savannah River Site was chemically separated and part was stored, costs would typically be between the boundaries for all-separation and all-storage.

Hybrid alternatives that ship about 1/4 of the foreign research reactor spent nuclear fuel to the United Kingdom Atomic Energy Authority's Dounreay facility and manage the remainder as in the U.S. chemical separation case generate minimum total program costs of about $650 million. Other cost factors would be about the same as in the chemical separation case.

At the level of accuracy in the costs presented here, alternatives based on chemical separation of aluminum-based spent nuclear fuel in the United States are likely to cost about the same as alternatives that divert a significant fraction of the spent nuclear fuel (aluminum-based and TRIGA) to Dounreay. Alternatives based on storage and direct disposal of spent nuclear fuel or some non-separation processing approach (e.g., melt and dilute) are likely to cost several hundred million dollars more.

F.1 Description of Existing and Proposed Technologies for Storage of Spent Nuclear Fuel

In this section, two major generic technologies will be presented. International and domestic types of each technology will be addressed. Section F.1.1 will discuss the dry storage designs available. Section F.1.2 will address wet storage technology types. The range of alternatives available to each site for the implementation of the proposed action is presented in Section F.1.3.

F.1.1 Dry Storage Designs

F.1.1.1 Overview of Dry Storage Approaches

There are several types of dry storage technology currently in use or proposed by various vendors at DOE sites as well as at commercial nuclear power facilities. These include:

- aboveground free-standing metal casks,
- aboveground free-standing concrete casks,
- aboveground free-standing dry storage buildings (vaults),
- inground lined and unlined holes or wells with or without casks,
- hot cells (buildings), and
- aboveground free-standing multi-purpose or dual-purpose casks.

It should be noted that additional support facilities for transfer and staging operations may be required in order to use the aforementioned dry storage technologies. A short discussion of the advantages and disadvantages of all dry storage technologies is given in the following sections.

It is important to appreciate the different approaches to handling weight and shielding. Today, most spent nuclear fuel facilities utilize a wet pool environment for handling, storing, and transferring spent nuclear fuel to transportation casks. The pool water provides shielding [usually a 3-meter (m) or 10-ft water cover is the minimum requirement], confinement of contamination, decay heat removal, and thermal capacity. All spent nuclear fuel elements weigh less than 0.9 metric tons (1 ton) and are readily moved within the pool by a crane of that capacity. Transportation containers (casks) for highway transport weigh between 18 and 36 metric tons (20 and 40 tons), and rail casks can weigh up to 91 metric tons (100 tons). Thus, most wet pool facilities have a bridge crane spanning the storage areas and the receiving bay(s) with a capacity of 45 to 91 metric tons (50 to 100 tons). Economical dry storage requires that a large number of elements be stored in each cask. Cask weights exceeding 91 metric tons (100 tons) are possible.

Dry storage manufacturers have overcome this problem by using metallic "transfer" canisters. These transfer canisters are considerably lighter than transportation casks, and usually weigh in the 9 to 27 metric tons (10 to 30 ton) range before loading. The transfer canister provides some shielding, but is principally for confinement of the spent nuclear fuel. They are loaded in the same manner as transportation casks.

For dry cask storage, the canister is loaded onto a truck and transferred to a previously constructed, shielded concrete cask away from the wet pool. With a vault, the canister is moved by a crane within a concrete shielded facility and placed in a storage tube within concrete shielding.

Radioactive materials in spent nuclear fuel require two levels of confinement for dry storage. These are usually the cladding material and the metal container (or transfer canister) within the metal cask or concrete structure (cask or vault). Leaking fuel elements can be dry stored provided they are placed within a separate metal container (i.e., "can") within the canister. This is relatively easy to accomplish, but can consume additional storage space. For foreign research reactor spent nuclear fuel, the impacts of canned fuel upon storage capacity should be minimal. The amount of canning expected for foreign research reactor spent nuclear fuel is not yet determined.

F.1.1.1.1 Aboveground Free-Standing Metal Casks

Metal storage casks are generally robust and some may even have been originally designed to meet transportation requirements. They are resistant to seismic loads, high winds, design basis tornado missiles, and accidental drops. The mechanism for heat removal is simple, using direct metal conduction to the external surface which is cooled by natural convection. They are not subject to air pathway blockage by snow, ice, or flooding. The shielding is accomplished by various means, primarily thick steel, lead, or cast iron wall sections. The dry metal casks are passive, requiring minimal surveillance. There are no high-temperature thermal limits on cask material; however, if the material is cast iron or ferritic steel, there may be low-temperature thermal limits to prevent brittle fracture. Brittle fracture is a phenomenon that occurs in some materials such as glass at normal temperatures, or in cast iron or some steels (ferritic) at low temperatures. Fracture requires a stress to initiate. Thermal limits always apply to the fuel cladding. This type of dry storage has a proven track record in the United States and overseas.

The disadvantages of the metal cask designs are the following. Frequently, metal cask designs are more expensive than concrete/metal hybrid designs or dry vault storage designs. The current metal cask designs use dual compressible "O" rings with a pressure gauge to monitor the confinement seal. "O" rings are gaskets which, when compressed, form a gas-tight seal. Seal leakage is a possible event which must be considered for this design. The metal cask may be very heavy, thus imposing a limiting factor for cranes at existing facilities.

F.1.1.1.2 Aboveground Free-Standing Concrete Casks

The advantages of concrete casks, as compared to all other storage technologies, are given below. Concrete cask systems are inexpensive relative to metal casks. The concrete casks require no active systems because they are totally passive. They consist of a welded cylindrical container or basket enclosing the spent nuclear fuel which is then placed inside either a vertical or horizontal concrete structure. The concrete shielding structure may be fabricated onsite. This type of dry storage has been utilized at commercial nuclear power plant facilities, for example: H.B. Robinson, Oconee, Calvert Cliffs, and Palisades. Concrete casks have also been licensed for use at the Brunswick plant. Many other utilities are already committed to taking this route for the interim storage of their commercial spent nuclear fuel.

The disadvantages for concrete cask systems are: (1) more surveillance is needed than with metal casks to verify no blockage of air passages, (2) they are not licensed for transportation over public roads, (3) they require a special purpose onsite shielded transportation cask, and (4) the long-term concrete temperature limit restricts the heat load of the spent fuel. However, for the foreign research reactor spent nuclear fuel, heat loads and fuel cladding temperature limits are a small fraction of the commercial spent nuclear fuel values. Therefore, high concrete temperatures are expected to be avoided.

F.1.1.1.3 Aboveground Free-Standing Dry Storage Building (Vault)

Vault storage consists of a large concrete aboveground building enclosing a vertical or horizontal array of spent nuclear fuel storage metal tubes and support systems, The advantages for the vault type of dry storage, as compared to all other storage technologies, are the following. For large quantities of spent nuclear fuel assemblies, the vault may have economic advantages when compared with either type of cask system. The heat removal is passive. The heat removal capacity for a properly designed vault is large, and therefore, there should be little concern for thermal limits being imposed (although there may be individual fuel decay heat limits). The vault which is licensed in the United States and abroad, has no high temperature limit associated with concrete. However, there is a low temperature limit because the secondary fuel confinement barrier is ferritic steel. To comply with current NRC 10 CFR 72 regulations, all spent nuclear fuel storage systems must have two confinement barriers. The intact fuel cladding is considered the first confinement barrier, and the cask or vessel is considered the secondary confinement barrier. The vault has a major advantage over all other types of dry storage because it provides a shielded means for loading the spent nuclear fuel on the vault premises. Another important advantage of the vault is the ease of spent nuclear fuel retrieval and monitoring while in storage. The vault includes facilities for inspection, placement in containers, and drying of wet fuel. The weight/volume of stored fuel is not a limiting factor. This type of system is currently in use at Fort St. Vrain in Colorado, at Wylfa, Wales, and is under construction at the PAKs nuclear power plant in Hungary.

The disadvantage is that, for small quantities of spent nuclear fuel, the cost may be higher than either the metal or concrete cask systems since a vault requires greater capital outlays.

F.1.1.1.4 Inground Lined and Unlined Wells With or Without Casks

The RISO National Laboratory's inground concrete block design relies on forced air convection heat transfer from the existing handling bay ventilation system, which includes High Efficiency Particulate Air filters and an air humidity monitoring system. Forced air heat removal is accomplished by directing the air around spent nuclear fuel containers and out through tubes embedded in the concrete. Like the pool storage systems, the RISO National Laboratory's system relies on active heat removal systems.

F.1.1.1.5 Hot Cell Facilities

Although hot cells are available at many facilities, including the Savannah River Site, the Idaho National Engineering Laboratory, and the Nevada Test Site, they can best be considered for small quantities of spent nuclear fuel for very short periods of time. Hot cells are basically set up to perform various operations on hazardous materials, and are generally not spacious enough to store materials on an indefinite or long-term basis. Furthermore, hot cells are frequently contaminated; this contamination may pose problems when it is time to transfer the foreign research reactor spent nuclear fuel to the repository. It is important to note that DOE possesses several unique hot cells that may be capable of foreign research reactor spent nuclear fuel storage due to their large, vault-like design.

F.1.1.1.6 Aboveground Free-Standing Multi-Purpose or Dual-Purpose Casks

The dual-purpose cask combines the functions of interim storage at a designated site and transportation on public roads, rail systems, or waterways. A multi-purpose cask may also add a third function of a repository canister; i.e., the cask and its contents need no further processing, characterization or identification in order to be compatible with the final repository. A dual-purpose or multi-purpose cask has attractive possibilities for the storage of spent nuclear fuel, regardless of the type of reactor (i.e., commercial or research reactor). Dual-purpose designs would satisfy the two functions of storage and transportation. For commercial utilities, this implies satisfaction of 10 CFR 71 requirements for transportation and 10 CFR 72 requirements for storage. By minimizing fuel handling operations, the dose for workers can be reduced, and the number of additional low-level waste products can be reduced. Minimization of fuel handling may also result in cost reductions, although this case has not been made. For a multi-purpose cask, satisfaction of 10 CFR 60 requirements is also necessary. DOE's Office of Civilian Radioactive Waste Management was actively pursuing a program to develop multi-purpose canister for domestic use (EG&G, 1994b; DOE, 1994f; DOE, 1994b; DOE, 1994c). However, DOE has decided in November 1995 to withdraw its proposal to prepare the EIS for this project.

The dual-purpose cask systems that are currently proposed offer a reduction in handling of the spent nuclear fuel in the storage to transportation operations, and multi-purpose cask systems offer an even greater reduction in handling in the operations involved at the repository site. However, no detailed cost/benefit analyses have been undertaken for foreign research reactor spent nuclear fuel. Furthermore, there is no basis at this time for concluding that either the "waste form" (intact spent nuclear fuel assemblies) or the sealed container will be compatible with the repository requirements. It is premature to draw any conclusion on the desirability to proceed with a multi-purpose cask system for foreign research reactor spent nuclear fuel use.

F.1.1.2 Specific Dry Storage Designs

There are no currently licensed dry storage systems specifically for foreign research reactor spent nuclear fuel in the United States. There are, however, many examples of dry spent nuclear fuel systems licensed by the NRC for commercial fuel. Table F-2 provides an overview of current manufacturers of dry storage systems. Table F-3 is a listing of dry storage systems currently licensed in the United States.

Dry storage systems must meet many design criteria, such as protection of fuel from degradation, shielding, thermal, criticality safety, structural integrity of confinement vessel, structural integrity of shielding, mechanical handling of fuel assemblies or canisters, containment and operational aspects. Some of these criteria are interrelated. For example, thermal criteria are designed to maintain fuel and cladding structural integrity. Shielding, thermal, and criticality parameters are the most important and are discussed in the following sections.

F.1.1.2.1 Shielding Design Comparisons

A spent nuclear fuel storage system must provide for adequate shielding of both the gamma and neutron radiation that emanate from irradiated nuclear fuel. The shielding must be designed to reduce the combined gamma and neutron dose rate to values that are below the limits for the public at the site boundary, collocated workers, and workers at the fuel storage facility. These limits are determined by Federal regulations such as 10 CFR 72 and 10 CFR 20. Shielding is designed for the maximum expected gamma and neutron source term, which is determined by performing computer code analyses of the nuclear fuel that account for the initial fuel fissile material inventory, its burnup in the reactor core, and the time after removal from the reactor (i.e., decay time) prior to its anticipated placement in the storage facility. The selection of a bounding and conservative set of these parameters results in the calculation of the highest possible gamma and neutron source term to be used in shielding design and analyses.

Shielding for gamma radiation relies on the use of high atomic weight or density materials, which attenuate and absorb gamma rays. The material selection depends on design limitations regarding shield thickness, cost, strength, and weight. The five materials which are almost always used in spent nuclear fuel storage facilities for gamma shielding are water, lead, steel, ductile iron or concrete. Lead and steel, having much higher densities and atomic weights than concrete and water, can provide relatively more

Table F-2 Dry Storage Technology Systems

Company	Metal Cask	Concrete Cask	Building (Vault)
AECL/Transnuclear	---	SILO	MACSTOR/CANSTOR
FW/GEC	---	---	MDV
GNS/GNSI	CASTOR	---	---
GNSI/OH	---	HDC	---
Nuclear Assurance Corp.	NAC	---	---
VECTRA	---	NUHOMS	---
Sierra Nuclear Corporation	---	VSC	---
RISO National Laboratory	---	DR3	---
Siemens Power Corporation	---	---	FUELSTOR
Transnuclear, Inc.	TN	---	---
Westinghouse Electric Corporation	MC-10	---	---
Atomic Energy of Canada, Ltd.	---	SILO	CANSTOR
Total Design:	**4**	**6**	**4**

FUELSTOR = Fuel Encapsulation and Lag Storage; FW/GEC = Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom); GNSI = General Nuclear Systems, Inc.; GNSI/OH = GNSI of Ontario Hydro; VSC = Ventilated Storage Cask; MDV = Modular Dry Vault

Table F-3 Dry Storage Systems Currently Licensed in the United States

Manufacturer	System	Location
General Nuclear Systems, Inc. (Chem-Nuclear)	CASTOR V/21	Surry
VECTRA	NUHOMS-7P, NUHOMS-24P, & NUHOMS-52B	Robinson, Oconee, & Calvert Cliffs
Westinghouse Electric Corporation	MC-10	Surry
Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom)	MDV	Wylfa, Wales UK PAKS (Hungary) Fort St. Vrain (USA)
Transnuclear	TN-24 & TN-40	Surry, Prairie Island
Sierra Nuclear Corporation	Ventilated Storage Cask-24	Palisades
Nuclear Assurance Corporation	NAC-C28, NAC-I28, NAC S/T, & NAC S/TC	Surry

effective gamma shielding with a smaller thickness of material. However, using steel and/or lead imposes a design penalty of increased cost. Concrete and water are much less expensive and may reduce overall shielding costs. It should also be noted that lead can be categorized as a Resource Conservation and Recovery Act waste, which restricts future decommissioning and disposal options. Concrete and water may also present unique safety problems, such as leakage (for water) or cracking (for concrete) during postulated accidents.

Neutron shielding requires low atomic weight materials because the uncharged neutron can only be absorbed by reducing its energy in collisions with nuclei similar in mass. Since the mass of a neutron is approximately one atomic mass unit, low atomic mass elements such as hydrogen, inert gas, lithium, carbon, and boron are suitable shields. Hydrogenous materials such as concrete and water are typically used in neutron shielding. It should be noted, however, that a sufficient thickness of heavier materials (such as high carbon steels) can provide neutron shielding. Also, shielding manufacturers offer products that have been artificially fortified in their hydrogen content, such as special forms of concrete, borated resin, and polyethylene. As in the case for gamma shielding, design factors in material selection include cost, density, weight, and safety.

The design of spent nuclear fuel storage facility shielding must also incorporate other factors along with cost, density, weight, and safety. Shielding usually performs a second function as a heat transfer medium from the spent nuclear fuel decay heat to the environment, and must therefore be able to effectively remove heat without exceeding fuel and shielding storage temperature limits. In some instances, the shielding also performs a structural function, either in handling or support.

Table F-4 shows a comparison of specific designs with a view toward shielding considerations. All of these designs will be discussed in more detail in subsequent sections of this appendix.

F.1.1.2.2 Thermal Design Comparisons

Spent nuclear fuel storage facilities are designed to effectively remove spent nuclear fuel decay heat during both incident-free operation and postulated accident conditions. Thermal design limits include long-term fuel storage cladding temperature to maintain cladding integrity and, in some cases, temperature limits of structural and/or shielding materials. Unlike pool storage systems, most of the dry storage systems emphasize passive heat removal. In contrast, active systems in wet pools include pumps, make-up water systems, filtration and water treatment systems, and heat exchangers.

All dry storage designs encapsulate the fuel, after it is dried, in a metal canister or tube that is evacuated (vacuum dried) and then filled with an inert gas such as helium. Helium is frequently used for its relatively high thermal conductivity that enhances heat conduction and heat transfer from the fuel to the encapsulating metal canister. Helium's inert properties also inhibit cladding corrosion. Since all the dry fuel storage technologies utilize a metal canister to enclose spent nuclear fuel, the first modes of heat transfer from the fuel to this canister's walls are heat conduction and radiation from the fuel cladding surface through the inert gas to the inside wall of the metal canister. Decay heat transfer from the encapsulating canister to the environment is accomplished by several different mechanisms dependent upon the specific storage design technology.

The dry metal cask design relies on its solid thick metal wall for conduction heat transfer from the fuel storage cavity to the atmosphere. Metal cask conduction heat transfer is not susceptible to any accident or degradation. This thermal design is inherently easy to analyze because conduction is a well-known heat transfer mechanism, and the thermal conductivity of such metal cask materials as carbon steel, stainless steel, and ductile cast iron is well known over the range of temperatures and conditions that are expected in the cask while storing spent nuclear fuel. With known design fuel decay heat, cask geometry (i.e., cask wall thickness), conduction material composition, and suitably conservative heat transfer assumptions from the cask metal surface to the ambient air, the temperature distribution within the cask and maximum fuel cladding temperature can be calculated with a high degree of certainty.

The dry concrete cask design uses a combination of conduction, natural convection, and radiation heat transfer to remove decay heat from the stored spent nuclear fuel and maintain acceptable operating temperatures. An air passageway around the storage canister is provided in this design because the relatively low thermal conductivity and allowable operating temperature limit of concrete, as compared to metal, prevent the concrete shield walls from serving as the primary means of decay heat removal. Radiation streaming requires that the inlet and outlet air passages to the cavity surrounding the canister be designed as a geometric labyrinth with suitable bends. One concrete cask design, the Atomic Energy of Canada, Ltd. SILO, does not have air passages but instead relies solely on conduction through solid

Table F-4 Comparison of Shield Design Parameters for Spent Nuclear Fuel Dry Storage Systems Currently Licensed in the United States

Manufacturer	Model	Shield Material	Shield Thickness	Design Limit Surface Dose Rate[a]
Nuclear Assurance Corporation	S/T	S.S., Lead, NS4FR	Radial: 20.3 cm (8 in) S.S. & 17.8 cm (7 in) NS4FR Axial: 12.7 cm (5 in) S.S.-PB & 7.6 cm (3 in) NS4FR	1 milliSievert/hr (100 mrem/hr)
Transnuclear, Inc.	TN-24	Borated Resin, C.S.	NA	Side: 0.57 milliSievert/hr (57 mrem/hr) Top: 0.11 milliSievert/hr (11 mrem/hr) Bottom: 0.45 milliSievert/hr (45 mrem/hr)
	TN-40	Borated Resin, C.S.	Radial: 21.6 cm (8.5 in) C.S. 11.4 cm (4.5 in) Resin Bottom: 22.2 cm (8.75 in) C.S. Top: 15.9 cm (6.25 in) Cast Iron	Side: 0.58 milliSievert/hr (58 mrem/hr) Top: 0.26 milliSievert/hr (26 mrem/hr) Bottom: 12.75 milliSievert/hr (1,275 mrem/hr)
Westinghouse Electric Corporation	MC-10	NS-3, C.S.	Radial: 25.4 cm (10 in) Steel, 7.6 cm (3 in) NS-3 Bottom: 25.4 cm (10 in) steel	2 milliSievert/hr (200 mrem/hr)
General Nuclear Systems, Inc.	CASTOR V21	Cast Iron, S.S., Polyethylene Rods	Radial: 30.5 cm (12 in) Bottom: 27.9 cm (11 in) Top: 39.1 cm (15.4 in) Rods Radial: 72-6.1 cm (2.4 in) Diameter	2 milliSievert/hr (200 mrem/hr)
VECTRA	NUHOMS 7P, 24P, and 52B	Concrete, S.S.	Side: 45.7/60.1 cm (18/24 in) Rear: 60.1 cm (24 in) Roof: 91.4 cm (36 in)	2 milliSievert/hr (200 mrem/hr) (at air inlet)
Sierra Nuclear Corporation	Ventilated Storage Cask-24	Concrete RX-277, Hydrogenated Concrete, C.S.	Radial: Steel & Concrete Top: RX-277 & Steel Bottom: Steel & Concrete	Side: 0.20 milliSievert/hr (20 mrem/hr) Top: 0.50 milliSievert/hr (50 mrem/hr) Air Inlet or Outlet: 1 milliSievert/hr (100 mrem/hr)
FW/GEC	MDV	Concrete	106.7 cm (42 in)	0.21 milliSievert/hr (21 mrem/hr)

NA = Not Available; C.S. = Carbon Steel; Pb = Lead; S.S. = Stainless Steel; NS-3 = Concrete; NS4FR = Special Fire-Resistant Castable Resin; RX-277 = Special Concrete with Extra Hydrogen; FW/GEC = Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom); MDV = Modular Dry Vault

[a] *These are limits established for commercial spent nuclear fuel assemblies. The dose rate expected from storage of foreign research reactor spent nuclear fuel is likely to be lower.*

concrete. The SILO's thermal design is acceptable only because it is limited to a much smaller total decay heat power than the air passage concrete casks.

Since a passive design is an underlying requirement of all dry concrete cask designs, the total airflow path from the cask air inlet to its outlet must include a sufficient elevation change to ensure natural convection airflow under all expected meteorological and heat load conditions.

The heat transfer from the canister follows two parallel paths: (1) convection from the surface of the canister to the naturally-induced airflow through the canister cavity, and (2) radiation and conduction heat transfer from the canister across the air in the cavity to the concrete shield and then conduction through the concrete shield wall thickness to the ambient air outside the concrete. Natural convection air heat removal is greater than the radiation and conduction through the air layer and concrete shield.

The heat transfer design of the concrete cask is vulnerable to accidents in which significant blockage of the air inlets and/or outlets restricts or prevents sufficient airflow into the canister cavity. Multiple inlets and outlets at different, and sometimes diametrically opposed, locations around the cask are used to reduce the likelihood of such an accident. Conservative adiabatic heatup analyses for these designs with commercial spent nuclear fuel have shown that temperature limits are not approached in more than 24 hours, even if the airflow inlets and outlets are completely blocked. Therefore, concrete cask sites have included a daily visual surveillance frequency for inspection of air inlets and outlets to ensure that they are not blocked. The adiabatic heatup for foreign research reactor spent nuclear fuel and its concomitant surveillance frequency may be different.

The concrete cask thermal design also requires more complex analyses for temperature distribution in both the fuel and the concrete due to the complex multidimensional and combined conduction-radiation-convection modes of heat transfer. An important thermal design issue for the concrete casks is proof that natural convection buoyancy-driven airflow will be induced through the inlet-cavity-outlet path under the entire range of expected wind and decay heat conditions, including the possibility of partial blockage that may be obscured from outside visual inspections. Unlike metal casks, which only have the fuel cladding temperature as a thermal limit, concrete casks are also limited by both the absolute magnitude and gradients of temperature within the concrete.

The concrete vault storage building represents a larger version of the concrete cask design in the realm of heat transfer. An array of vertically or horizontally oriented metal canisters enclosing spent nuclear fuel is surrounded by a concrete building with labyrinth air inlet and outlet passages. With the exception of size, this design utilizes the same modes of heat transfer as the concrete cask. Its inherently larger flow areas for inlets and outlets and typically larger elevation from inlet to outlet provide a greater natural convection airflow and reduce vulnerability to airflow passage blockage.

Specific Thermal Features

Thermal design performance parameters of specific manufacturers' dry storage technologies are presented in Table F-5. This table shows that all dry spent nuclear fuel storage technologies use radiation and conduction as heat transfer mechanisms, and that concrete-based systems also rely on internal air passage natural convection heat transfer. All the systems have fuel cladding temperature limits, but systems relying on concrete also have concrete temperature limits.

Pool storage systems utilize an active cooling system with pumps and heat exchangers that remove decay heat transferred to the pool water from stored fuel via conduction and natural convection. The relatively large mass and heat capacity of the pool water provide a significant margin of time before the pool water reaches its boiling temperature in the event of a cooling system failure.

The RISO National Laboratory's inground concrete block design relies on forced air convection heat transfer from the existing handling bay ventilation system, which includes High Efficiency Particulate Air filters and an air humidity monitoring system. Forced-air heat removal is accomplished by directing the heating, ventilation, and air conditioning air around the stored fuel, and then out through separate tubes embedded in the concrete. Like the pool, the RISO National Laboratory's system relies on active heat removal systems.

The Atomic Energy of Canada, Ltd., SILO is an exception to the previously discussed concrete cask designs because it relies solely on conduction through a solid concrete structure for decay heat removal,

Table F-5 Comparison of Thermal Design Parameters for Spent Nuclear Fuel Dry Storage Systems Currently Licensed in the United States

Manufacturer	Model	Design Heat Load (KW)	Thermal Limits[a] (°C)	Heat Transfer Mode(s)[b]
Nuclear Assurance Corp.	S/T	26	NA	Conduction, Radiation
	S/TC	22	NA	Conduction, Radiation
Transnuclear, Inc.	TN 24	24	149 - Resin	Conduction, Radiation
	TN 40		NA	Conduction, Radiation
Westinghouse Electric Corp.	MC-10	13.5	340 - LWR Cladding	Conduction, Radiation
GNSI	CASTOR V21	21	370 - LWR Cladding	Conduction, Radiation
VECTRA	NUHOMS 7P and 24P	24	340 - Fuel Clad Normal 570 - Fuel Clad Accident	Conduction, Radiation, Natural Convection
	Standardized 24P and 52B	24 and 19	225 - Concrete Accident 177 - Concrete Normal 578 - Fuel Clad Accident 378 - Fuel Clad Normal	Conduction, Radiation, Natural Convection
Sierra Nuclear Corporation	Ventilated Storage Cask-24	24	93 - Concrete Normal 177 - Concrete Accident 570 - Fuel Clad Accident 378 - Fuel Clad Normal	Conduction, Radiation, Natural Convection
FW/GEC	Modular Dry Vault	0.15 per HTGR[c] Canister	399 - for Fort St. Vrain Type Fuel	Conduction, Radiation, Natural Convection

NA = Not Available; LWR = Light Water Reactor; FW/GEC = Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom); GNSI = General Nuclear Systems, Inc.

[a] *Fuel Limits are for Commercial Light Water Reactor Zircaloy Clad Fuel Type or for HTGR fuel.*

[b] *Heat transfer modes are for commercial spent nuclear fuel.*

[c] *HTGR= High Temperature Gas Reactor Type Fuel from Fort St. Vrain.*

without internal natural convection airflow around the canister. The SILO can maintain acceptable concrete and fuel cladding temperatures without internal airflow passages by limiting its contained total fuel heat load to about 4 kilowatts, as compared to the 24 kilowatts typical of other concrete casks with airflow passages. This lower heat load may not be limiting for the storage of foreign research reactor spent nuclear fuel since it does not produce decay heat as high as that for commercial nuclear power plant fuel for the same decay time.

F.1.1.2.3 Criticality Prevention Design of Spent Nuclear Fuel Storage Technology

A self-sustaining nuclear fission process is called criticality. Unlike the previously discussed thermal and shielding designs, criticality prevention design for spent nuclear fuel storage facilities does not rely on materials outside of the fuel storage basket or canister. Instead, the canister interior fuel support structure and fuel specifications for storage are the determining factors in criticality control.

Spent nuclear fuel storage facilities are shown to meet specific regulatory subcriticality requirements by conservative criticality analyses. These analyses conservatively assume that the spent nuclear fuel has its original enrichment of fissile material [e.g., fresh unirradiated fuel weight percent uranium-235 (^{235}U)]. In reality, the fuel has been irradiated and the initial concentration of fissile material is reduced from its original value through fission reactions producing numerous fission products.

Suitable criteria for establishing nuclear criticality safety have been documented (ANSI, 1984b, 1983, and 1975b). These documents deal specifically with, respectively, the storage of commercial spent nuclear fuel outside of the reactor and in dry storage installations.

Another conservative aspect of these criticality analyses is the requirement that a sensitivity study be performed that varies the water concentration within the canister free volume from no water to 100 percent water, to optimize moderation density. These analyses usually show that the most reactive (i.e., closest to critical conditions) configuration occurs with a water density less than that equivalent to a fully flooded canister (related to enrichment).

The criticality analyses explicitly model the fuel geometry, all materials present in the fuel, and the structural spacer design within the canister. Center-to-center distance for the fuel in the canister is another important parameter in determining the reactivity of the stored fuel.

In summary, the criticality prevention design of spent nuclear fuel storage facilities ensures that each canister will remain subcritical throughout the entire operation, during both incident-free and accident conditions. The criticality prevention design incorporates the following features:

- fuel specifications, including type of fuel, maximum initial fresh fuel ^{235}U enrichment, and number of fuel assemblies to be stored in a single canister,
- fuel assembly spacing inside the canister as set by the presence of structural support and spacing members, and
- the presence and composition of any neutron absorbing material between adjacent fuel assemblies inside the canister.

F.1.1.2.4 Current NRC-Licensed, Dry Storage Technologies for Commercial Spent Nuclear Fuel

The technologies discussed in this section are described in terms of their use for storage of commercial spent nuclear fuel. Foreign research reactor spent nuclear fuel storage design parameters will be different for each technology.

F.1.1.2.4.1 Nuclear Assurance Corporation S/T, NAC-C28 S/T, NAC-I28

Description of Nuclear Assurance Corporation S/T, NAC-C28 S/T and NAC-I28

Two of the Nuclear Assurance Corporation metal casks for the storage of spent nuclear fuel are in use at the Surry Nuclear Power Plant in Virginia. The Nuclear Assurance Corporation S/T design uses a combination of stainless steel and lead for gamma shielding and NS4FR, which is a fire-resistant castable resin, for neutron shielding (NRC, 1988a). To ensure a surface contact dose rate of less than 100 mrem/hr, 20.3 cm (8 in) of stainless steel and lead and 17.8 cm (7 in) of NS4FR are used in the cylindrical wall, while the top and bottom shields are composed of 7.6 cm (3 in) of NS4FR and about 12 cm (5 in) of steel and lead. Total weight of the loaded cask is either 91 metric tons (100 tons) for 26 intact Pressurized Water Reactor fuel assemblies, or 112 metric tons (124 tons), which is just under the 125-ton limit of many loading cranes for the 56 consolidated fuel assembly model. The Nuclear Assurance Corporation S/T models have been licensed by the NRC. The Nuclear Assurance Corporation S/T is shown in Figure F-2.

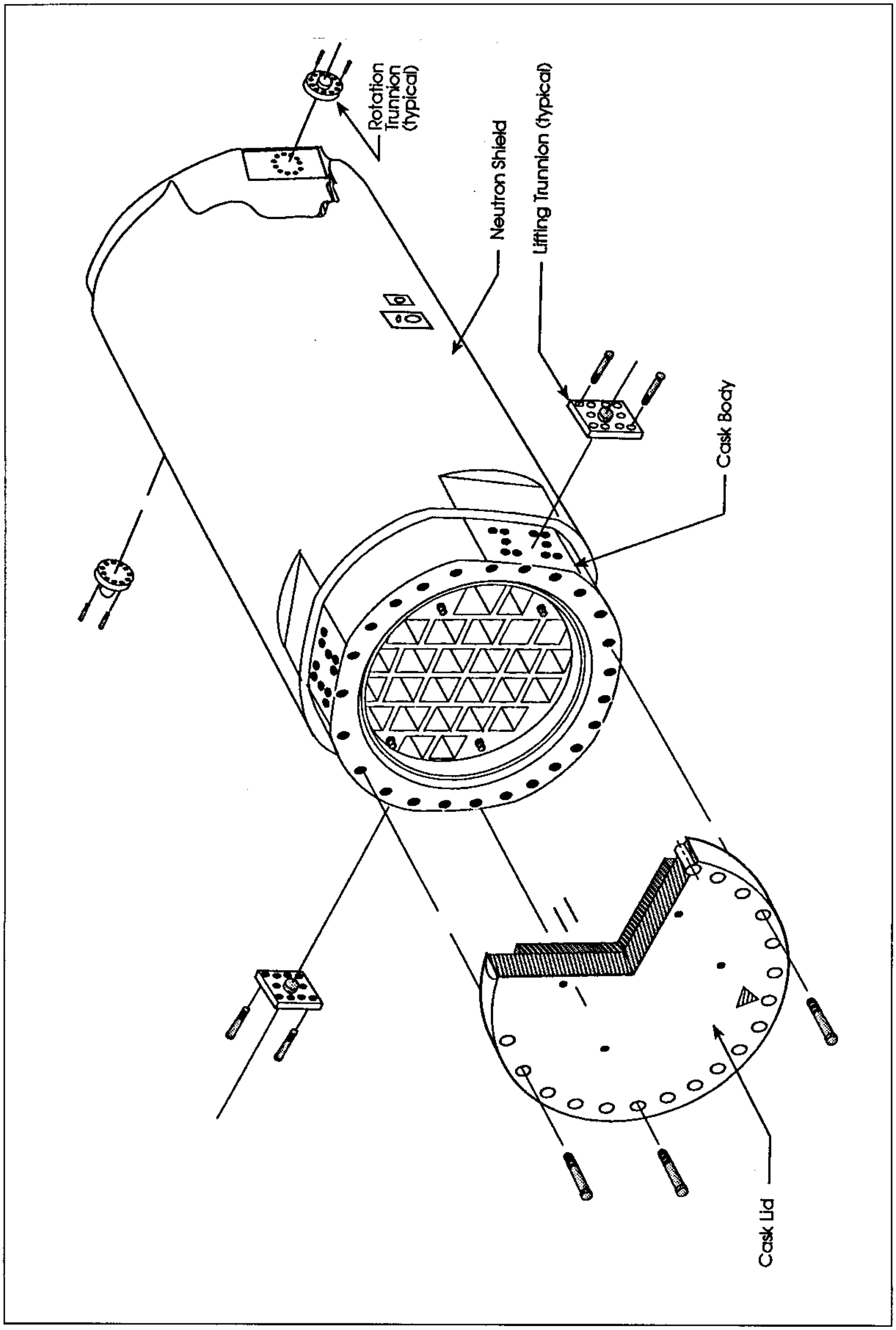


Figure F-2 Nuclear Assurance Corporation Spent Nuclear Fuel Storage Cask Body NAC S/T

NRC Certification or Basis for License

The NRC has granted a Certificate of Compliance to Model Nuclear Assurance Corporation S/T (Certificate Number 1002). Nuclear Assurance Corporation Model NAC-C28 S/T is also certified with Certificate Number 1003. The basis for these certificates is 10 CFR 72 Subparts K and L. Nuclear Assurance Corporation model NAC-I28 is currently licensed on a site-specific basis at Surry Nuclear Power Plant based on 10 CFR 72 Subparts A through I.

F.1.1.2.4.2 General Nuclear Systems, Inc. CASTOR V/21

Description of General Nuclear Systems, Inc. CASTOR V/21

In the United States, the General Nuclear Systems, Inc. CASTOR V/21 has been approved by the NRC and is in use at the Surry Nuclear Power Plant. This design relies on thick ductile cast iron and polyethylene as both its gamma and neutron shields. Ductile cast iron contains significant quantities of nodular graphite, which is essentially carbon, a good neutron shield. Polyethylene is a form of plastic that is high in hydrogen. The ductile cast iron shield is 30.5 cm (12 in) thick. Additional neutron shielding is provided by seventy-two 6.1 cm (2.4 in) diameter polyethylene rods placed in axial holes in the cast iron wall. The top lid shielding is 39.1 cm (15.4 in) of stainless steel, and the bottom lid shielding is 27.9 cm (11 in) of ductile cast iron. The V/21, holding 21 Pressurized Water Reactor fuel assemblies at Surry, weighs 96 metric tons (106 tons) fully loaded. A sketch of the CASTOR V/21 is presented in Figure F-3. The shielding design basis is for a surface contact dose rate less than 200 mrem/hr. There is a wide range of CASTOR designs for a variety of fuel types, including test reactor fuel. A conceptual design [CASTOR Material Test Reactor (MTR) 2] for a dual-purpose, transport/storage cask for research reactor fuel has been developed. This cask uses the same basic ductile cast iron body for shielding.

NRC Certification or Basis for License

The NRC has granted Certificate of Compliance Number 1000 for the General Nuclear Systems, Inc. model CASTOR V/21 under the terms of 10 CFR 72 Subparts L and K (Models X/28 and X/33 are not currently licensed, but are being reviewed by the NRC).

F.1.1.2.4.3 Westinghouse Electric Corporation MC-10

Description of Westinghouse Electric Corporation MC-10

The Westinghouse Electric Corporation MC-10 metal cask has been approved by the NRC and is in use at the Surry Nuclear Power Plant site (NRC, 1987). This cask design utilizes thick carbon steel and BISCO NS-3 hydrogenated concrete for shielding. The NS-3 provides neutron shielding, while the carbon steel is used for gamma shielding. Total radial neutron and gamma shielding is approximately 33 cm (13 in), while axial shielding is about 25.4 cm (10 in). The design surface contact dose rate is 200 mrem/hr, which bounds the actual vendor-calculated maximum surface contact dose rates of 7, 38, and 57 mrem/hr at the top, side, and bottom of the cask. The MC-10 was designed to hold 24 Pressurized Water Reactor fuel assemblies and weighs 103 metric tons (113.3 tons) fully loaded.

NRC Certification or Basis for License

The NRC has issued Certificate of Compliance Number 1001 for the metal cask model MC-10 in accordance with the terms of 10 CFR 72 Subparts L and K.

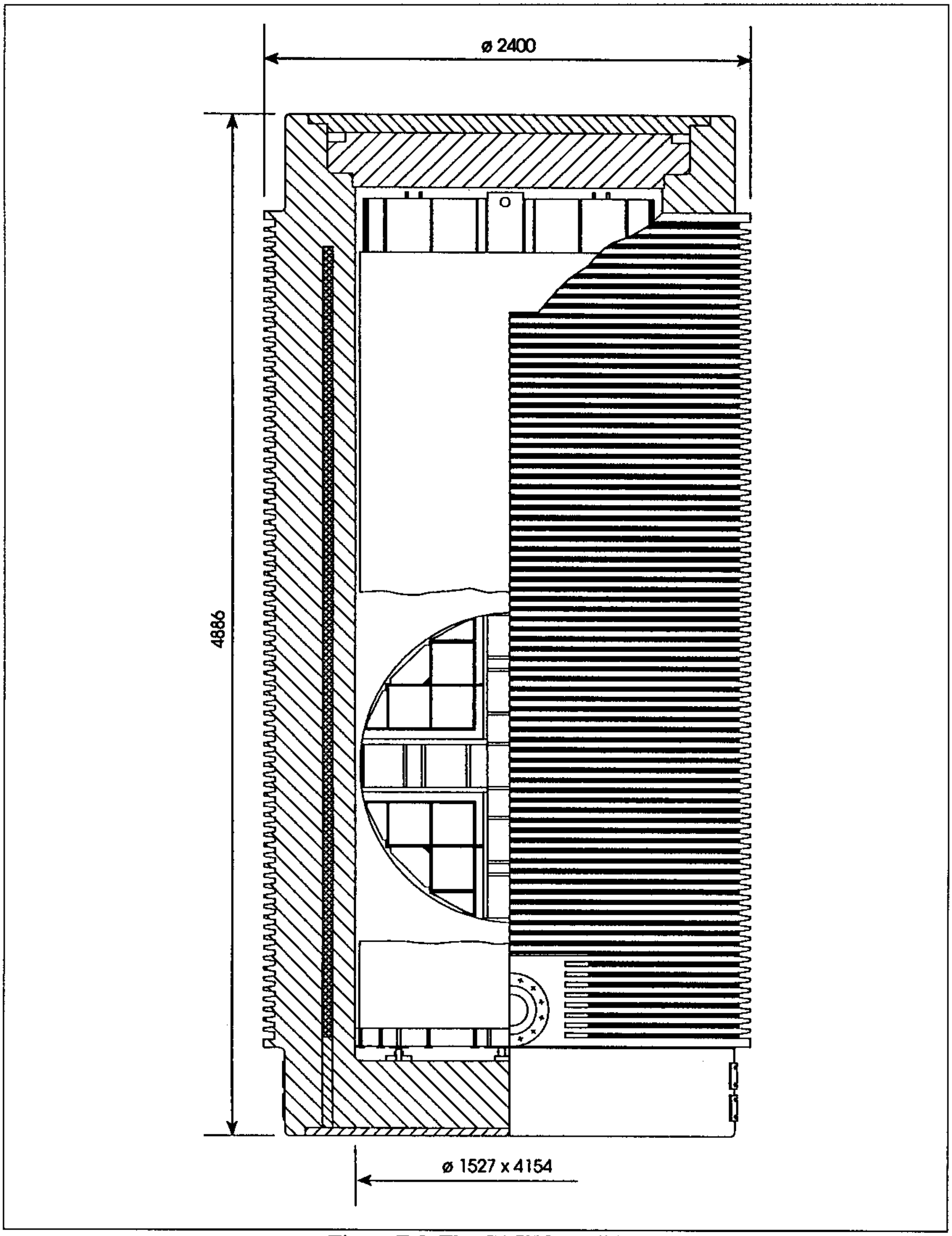


Figure F-3 The CASTOR V/21

F.1.1.2.4.4 Transnuclear, Inc. TN-24 and TN-40

Description of Transnuclear, Inc. TN-24, TN-40

The Transnuclear, Inc. design has been developed and produced for a large number of storage and transportation systems for radioactive materials, including spent nuclear fuel. The TN-24 and TN-40 models store 24 and 40 spent Pressurized Water Reactor fuel assemblies, respectively. TN systems feature metal casks for both transportation and storage of spent fuel. The TN-24 is an NRC-licensed storage cask that uses carbon steel for gamma shielding and a borated resin for neutron shielding (NRC, 1989). The TN-40 is a newer model that uses a two-metal shell design, with the inner shell consisting of high quality carbon steel for containment and the outer shell providing shielding and heat transfer, but of a lower quality steel. For the two models, top and side contact dose rate limits are less than 100 mrem per hour, but the bottom of the cask may have a contact dose rate limit as high as 1,275 mrem/hr. It should be noted that the normal configuration for these casks is to be standing upright on their bottoms, thereby precluding exposure to this relatively higher dose rate. A sketch of a Transnuclear, Inc. TN cask is shown in Figure F-4.

NRC Certification or Basis for License

The TN-24 model has been issued NRC Certificate of Compliance Number 1005 and is licensed according to 10 CFR 72 Subparts L and K. The TN-40 model is licensed on a site-specific basis at the Prairie Island Nuclear Power Plant in Minnesota (owned by Northern States Power) under the provisions of 10 CFR 72 Subparts A through I. The Transnuclear, Inc. Model TN-32 is not yet approved.

F.1.1.2.4.5 VECTRA Design NUHOMS-7P, -24P, and -52B

Description of VECTRA NUHOMS-7P, -24P, and -52B

VECTRA's NUHOMS designs utilize a horizontal concrete dry storage system for spent nuclear fuel (NUTECH, 1988). The NUHOMS-7P and NUHOMS-24P designs have been approved by the NRC for Pressurized Water Reactor spent nuclear fuel and are in use at the Robinson, Oconee, and Calvert Cliffs Nuclear Power Plant sites. The NRC approved the use of NUHOMS-52B for the Brunswick power plant, but the utility shipped this spent nuclear fuel to its Robinson plant. The NUHOMS design uses concrete as both gamma and neutron shielding. The requirement for internal air passages to allow natural convection heat removal from the metal storage canister placed within the concrete structure required 90 degree bends in the concrete shield for air passages to avoid radiation streaming and more detailed shielding analyses. The reinforced side wall concrete shield thickness is 45.7 or 61 cm (18 or 24 in) depending on location in the array, while the rear wall is 61 cm (24 in) thick and the roof is 91.4 cm (36 in) thick. The maximum surface contact dose rate limit at the air inlet is 200 mrem/hr. A sketch of the NUHOMS-24P system is shown in Figure F-5.

NRC Certification or Basis for License

NUHOMS models 7P and 24P are licensed at specific sites under the provisions of 10 CFR 72. Vectra has also received a license from the NRC for their standardized NUHOMS-24P and -52B models for use by the light water reactor utilities.

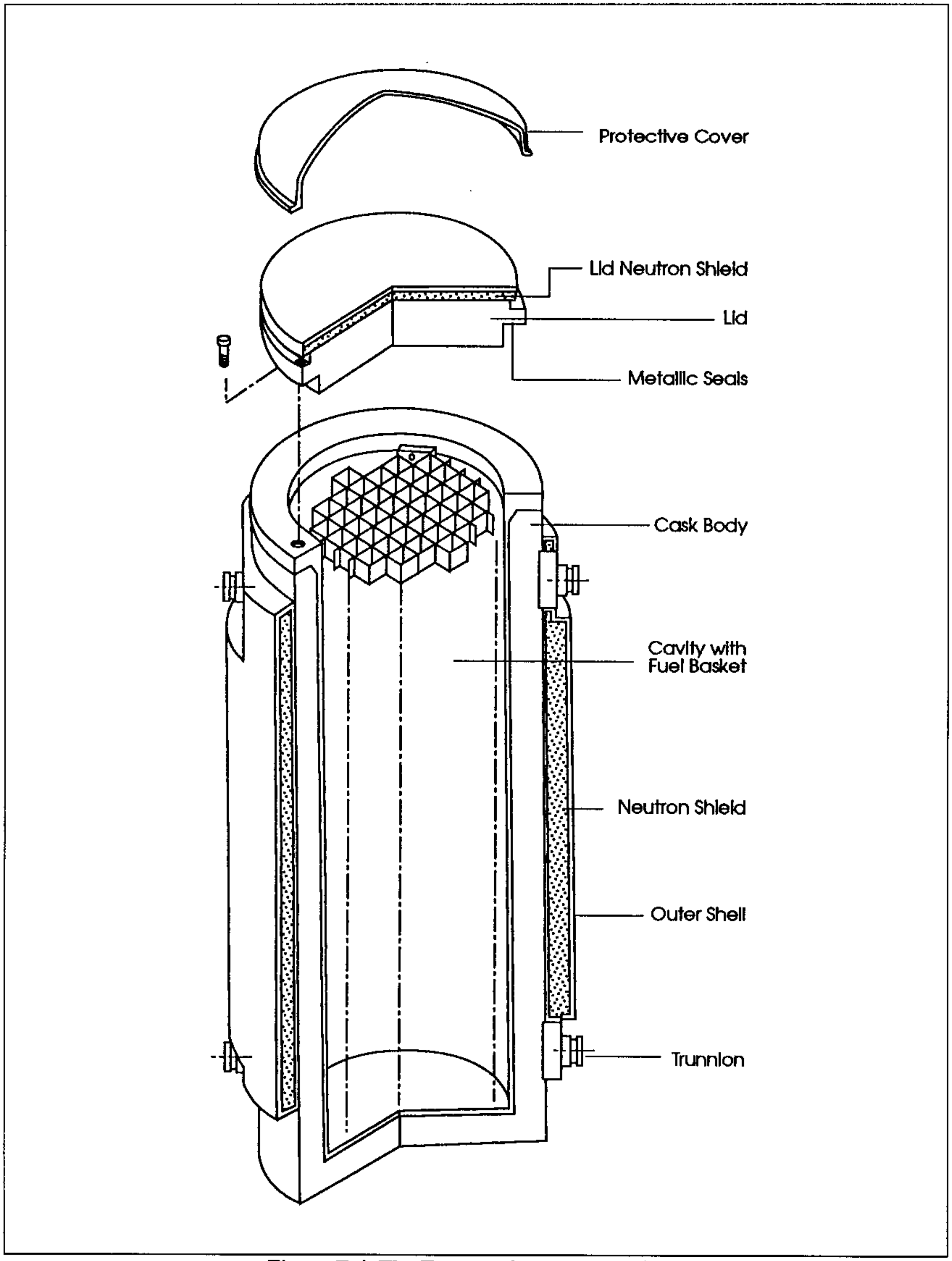


Figure F-4 The Transnuclear, Inc. TN Cask

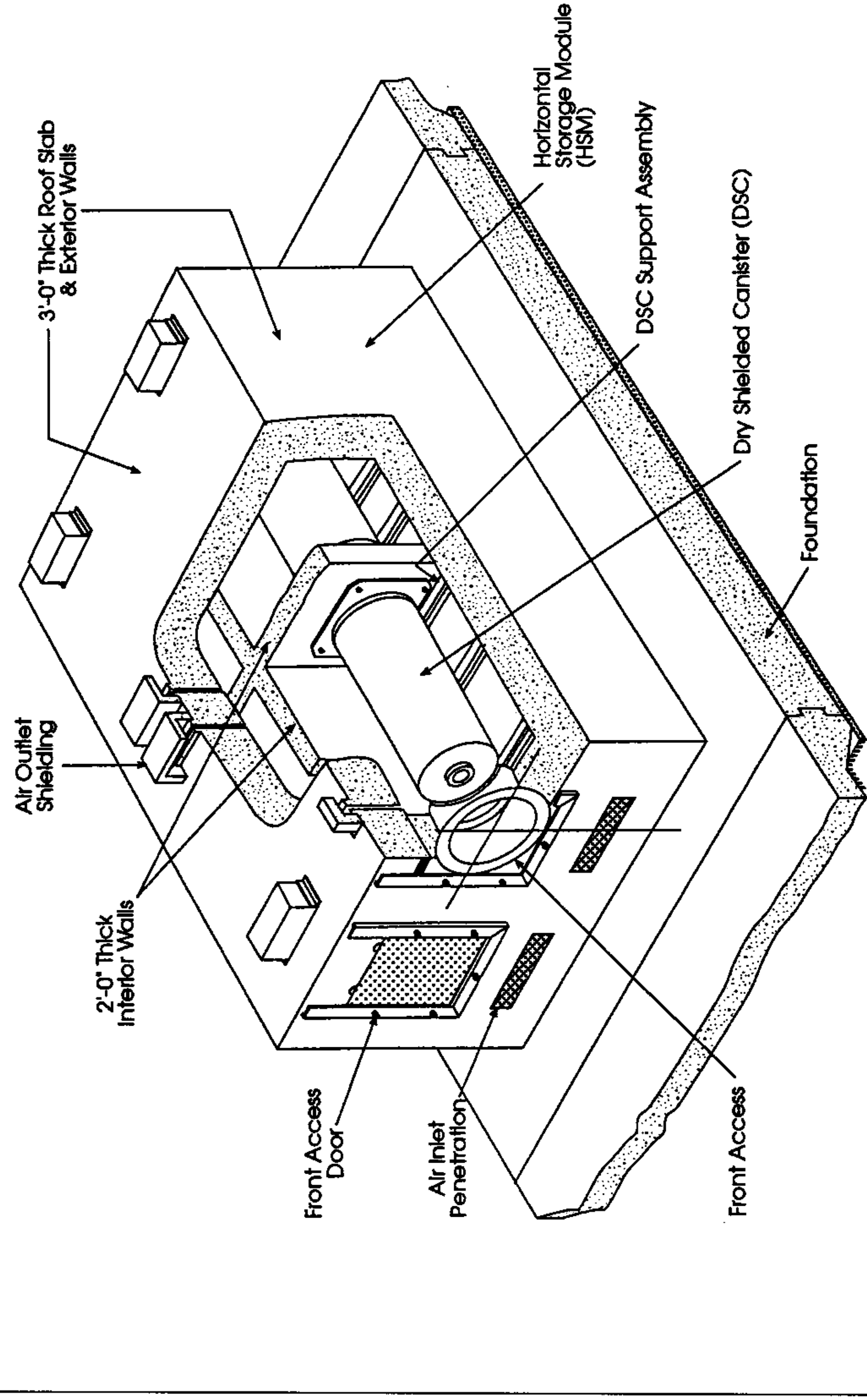


Figure F-5 The NUHOMS-24P Horizontal Storage Module Components

F.1.1.2.4.6 Modular Dry Vault

Description of Modular Dry Vault

The modular dry vault spent nuclear fuel storage system [designed by Foster Wheeler/GEC Alsthom Engineering Systems, Ltd. (United Kingdom)] is the only vault system in the United States that has been approved by the NRC and is in operation at the Fort St. Vrain nuclear power plant site. The modular dry vault places spent nuclear fuel in vertically oriented cylindrical steel fuel storage containers which are then inserted into a steel charge face structure within the thick concrete structure. A labyrinth airflow passage system provides natural convection airflow for decay heat removal. The shielding is provided by the 106.7 cm (42 in) thick concrete walls and the labyrinth airflow passages. For the Fort St. Vrain fuel, maximum design modular dry vault surface dose rate is 21 mrem/hr. A picture of the cross section of the modular dry vault is shown in Figure F-6.

NRC Certification or Basis for License

The modular dry vault model has been approved by the NRC for the site-specific application at Fort St. Vrain. The basis for the license is 10 CFR 72.

F.1.1.2.4.7 Ventilated Storage Cask System (VSC-24)

Description of VSC-24

The Ventilated Storage Cask, designed by Sierra Nuclear Corporation, is a vertical concrete cask design that has been approved by the NRC and is in use at the Palisades nuclear power plant site. As with the NUHOMS design, this system relies on concrete for both neutron and gamma shielding and incorporates internal airflow passages requiring detailed shielding analyses to demonstrate acceptable streaming doses. The Ventilated Storage Cask design dose rates are 20 mrem/hr side contact and 50 mrem/hr top contact. A sketch of the Ventilated Storage Cask is shown in Figure F-7.

NRC Certification or Basis for License

The Sierra Nuclear Corporation's Model VSC-24 has been granted Certificate of Compliance Number 1004 by the NRC. The basis for this certificate is 10 CFR 72 Subparts L and K.

F.1.1.2.5 Manufacturers of Commercial Nuclear Fuel Dry Storage Systems Not Currently Licensed by the NRC in the United States

In addition to the above examples of dry cask storage systems licensed in the United States, there are other systems either licensed outside the United States or in the design and/or licensing stage (Table F-6). Tables F-7 and F-8 show shielding and thermal related parameters of the various dry cask models that are not currently licensed in the United States.

F.1.1.2.5.1 Description of MACSTOR

The MACSTOR system (designed by Atomic Energy of Canada, Ltd. and Transnuclear, Inc.), representing a synthesis of both metal and concrete casks in a modular dry vault, is being reviewed for use in Canada (AECLT, 1994). Spent nuclear fuel is placed in 0.95 cm (0.375 in) thick carbon steel canisters or baskets that are then placed (in a vertical position) in concrete modules. Air labyrinth passages into and

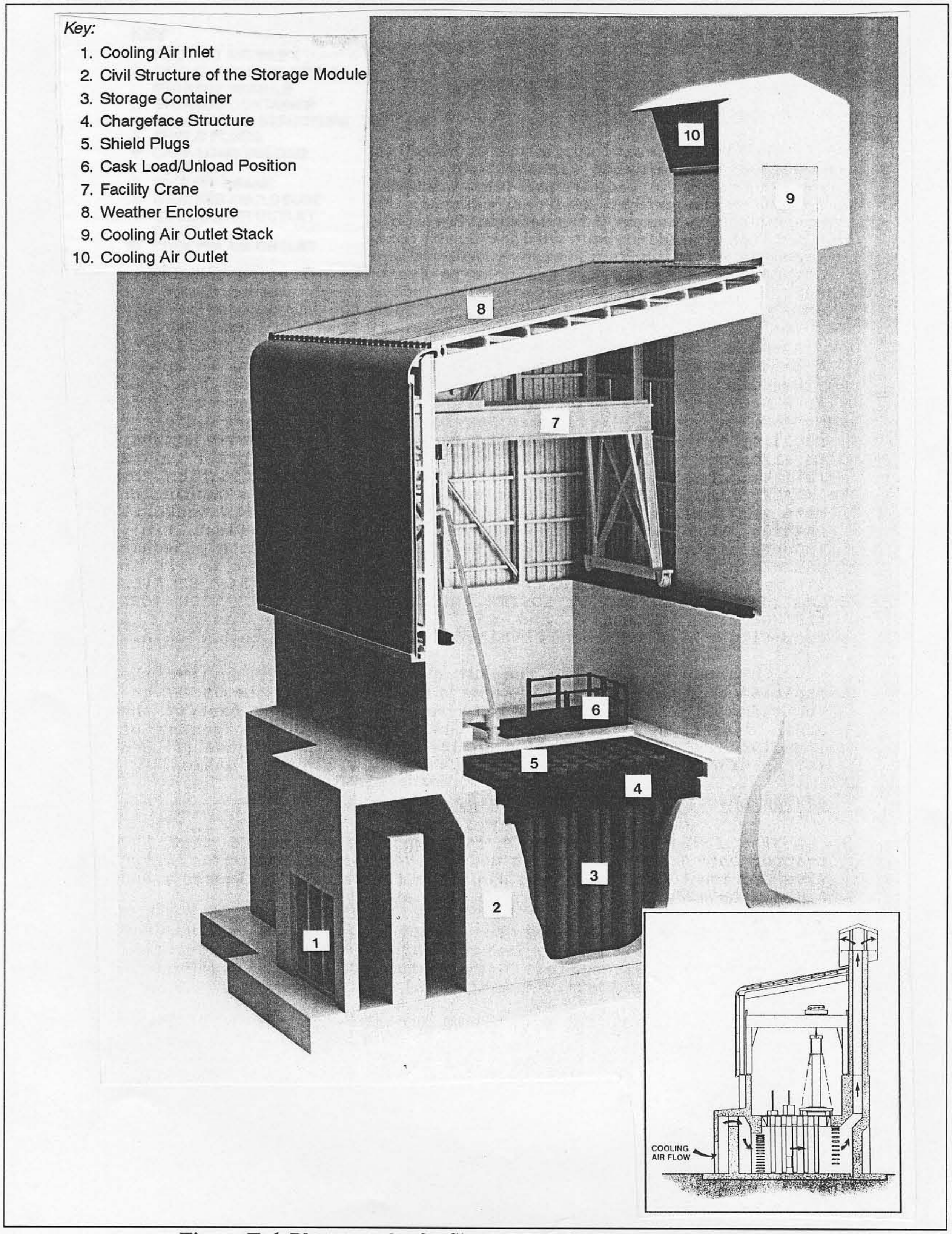


Figure F-6 Photograph of a Single Modular Dry Vault Module

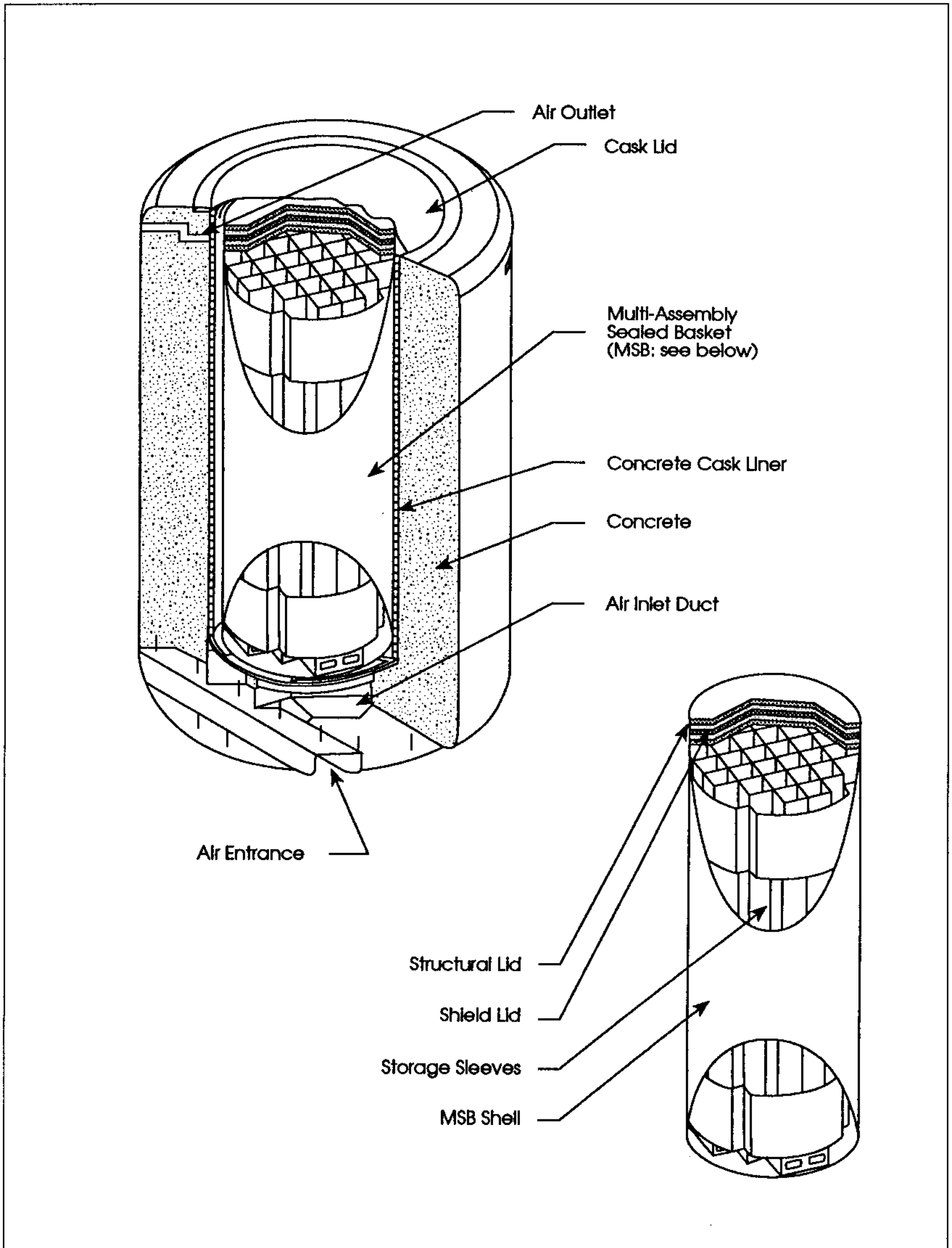


Figure F-7 Ventilated Storage Cask System Components

Table F-6 Manufacturers of Dry Storage Systems Not Currently Licensed in the United States

Manufacturer	Facility or Model	Location or Status
General Nuclear Systems, Inc. of Ontario Hydro	HDC	Canada
Atomic Energy of Canada, Ltd./Transnuclear Inc.	MACSTOR	Canada
Siemens Power Corporation	FUELSTOR	Germany
RISO National Laboratory	DR3	Denmark
Atomic Energy of Canada, Ltd.	SILO, Canister	Canada
Nuclear Assurance Corporation	NAC-26	10 CFR 72 License Pending
General Nuclear Systems, Inc.	X28 and X33	10 CFR 72 License Pending
Transnuclear, Inc.	TN-32	10 CFR 72 License Pending

FUELSTOR = Fuel Encapsulation and Lag Storage

Table F-7 Comparison of Shield Design Parameters for Spent Nuclear Fuel Dry Storage Systems Not Currently Licensed in the United States

Manufacturer	Model	Shield Material	Shield Thickness in cm (in)	Design Limit Surface Dose Rate milliSievert/hr (mrem/hr)
GNSI/OH	HDC-36	High Density Carbon Steel, Concrete	2.5 (1) CS, 45.7 (18) High-Density Concrete	NA
Siemens Power Corporation	FUELSTOR	Concrete	304.8 (120)	0.00001 (0.001)
Transnuclear/Atomic Energy of Canada Ltd.	MACSTOR	Concrete	96.5 (38)	0.025 (2.5)
RISO National Laboratory	DR3	Carbon Steel, Concrete, Earth	76.2 (30) Carbon Steel, Axial 15.2 (6) Concrete, Radial	NA
Atomic Energy of Canada Ltd.	SILO	Concrete	91.4 (36) Concrete	0.025 (2.5)
Atomic Energy of Canada Ltd.	CANSTOR	Concrete	NA	NA
Transnuclear	TN-32	Borated Resin Carbon Steel	Radial: 21.6 (8.5) 11.4 (4.5) Resin Bottom: 22.2 (8.75) Carbon Steel Top: 15.9 (6.25) Cast Iron	Side: 0.86 (86) Top: 0.18 (18) Bottom: 3.15 (315)
General Nuclear Systems, Inc.	CASTOR X28 and X33	Cast Iron, Stainless Steel Polyethylene Rods	Radial: 30.5 (12) Bottom: 27.9 (11) Top: 39.1 (15.4) Rods Radial: 72- 6.1 (2.4) Dia.	2 (200)

NA = Not Available; FUELSTOR = Fuel Encapsulation and Lag Storage ; GNSI/OH = General Nuclear Systems, Inc. of Ontario Hydro; Transnuclear = Transnuclear, Inc.

out of the module provide a flow path for natural convection airflow to remove decay heat. The vault concrete walls are 96.5 cm (38 in) thick and designed to reduce dose rates to less than 2.5 mrem/hr on contact. This concrete thickness is maintained even where the airflow passage labyrinth is located. A cross section of the MACSTOR is given in Figure F-8.

F.1.1.2.5.2 Description of a Fuel Encapsulation and Lag Storage Facility

The Fuel Encapsulation and Lag Storage system is similar to the modular dry vault and MACSTOR/CANSTOR in that it is a stand-alone concrete building with interior steel storage containers. Unlike the modular dry vault and CANSTOR/MACSTOR, the Fuel Encapsulation and Lag Storage system

Table F-8 Comparison of Thermal Design Parameters for Spent Nuclear Fuel Dry Storage Systems Not Currently Licensed in the United States

Manufacturer	Model	Design Heat Load (KW)	Thermal Limits (°C)	Heat Transfer Mode(s)
GNSI/OH	HDC	---	---	Conduction, Radiation
Siemens Power Corporation	FUELSTOR	Up to 2 KW Per Canister	380/2 KW 250-1KW-Clad	Conduction, Radiation, Natural Convection
Transnuclear Inc./Atomic Energy of Canada Ltd.	MACSTOR CANSTOR	240 (20 Canisters at 12 each)	93 Concrete 340 LWR Clad	Conduction, Radiation
RISO National Laboratory	DR3	---	NA	Conduction, Radiation, Forced Convection
Atomic Energy of Canada Ltd.	SILO	4	NA	Convection
Transnuclear Inc.	TN-32	---	NA	Conduction, Radiation
General Nuclear Systems, Inc.	CASTOR X28 and X33	19.2 and 20.9	370 LWR Cladding	Conduction, Radiation

NA = Not Available; LWR = Light Water Reactor; FUELSTOR = Fuel Encapsulation and Lag Storage; GNSI/OH = General Nuclear Systems, Inc. of Ontario Hydro

stores spent nuclear fuel containers in a horizontal position. The Fuel Encapsulation and Lag Storage system is designed for a surface contact dose rate of 0.001 mrem/hr, and relies on its combined 304.8 cm (120 in) thick inner and outer shield concrete walls and labyrinth airflow passages for shielding. The Fuel Encapsulation and Lag Storage system is not licensed by the NRC or in use in the United States A cross section of the Fuel Encapsulation and Lag Storage system is shown in Figure F-9.

F.1.1.2.5.3 Description of a RISO National Laboratory Facility

In Denmark, the RISO National Laboratory has designed and constructed a dry storage facility at its DR3 PLUTO type research reactor to store MTR spent nuclear fuel from this reactor. This facility was installed under the floor of the active handling bay at the reactor and consists of four prefabricated octagonal concrete blocks placed in a vertical position into steel lined holes in the earth. Each block contains 12 storage holes in a triangular mesh, with a carbon steel form forming each hole and a separate stainless steel tube containing the spent nuclear fuel. Axial shielding is provided by a 76.2 cm (30 in) thick carbon steel plug. Radial shielding is provided by the surrounding earth and concrete of the octagonal block with the minimum concrete thickness of 15.2 cm (6 in). A sketch of the RISO National Laboratory's design is shown in Figure F-10.

F.1.1.2.5.4 Description of SILO

The SILO has been designed and licensed in Canada, and over 180 concrete SILOs have been built for the storage of Canadian research reactor and CANDU-commercial reactor spent fuel (AECLT, 1994). The SILO consists of a carbon steel-lined cylindrical hole inside a 91.4 cm (36 in) thick vertical concrete cylinder without any labyrinth airflow passage for heat removal. Carbon or stainless steel canisters containing the spent nuclear fuel are placed inside the SILO and stacked up to nine high before being covered by a steel and concrete plug. The surface SILO dose rate limit is 2.5 mrem/hr. The unique design aspect of the SILO is that it is the only concrete cask without airflow passages for natural convection heat removal. It has been used for short-length low decay heat fuel which is dimensionally similar to foreign

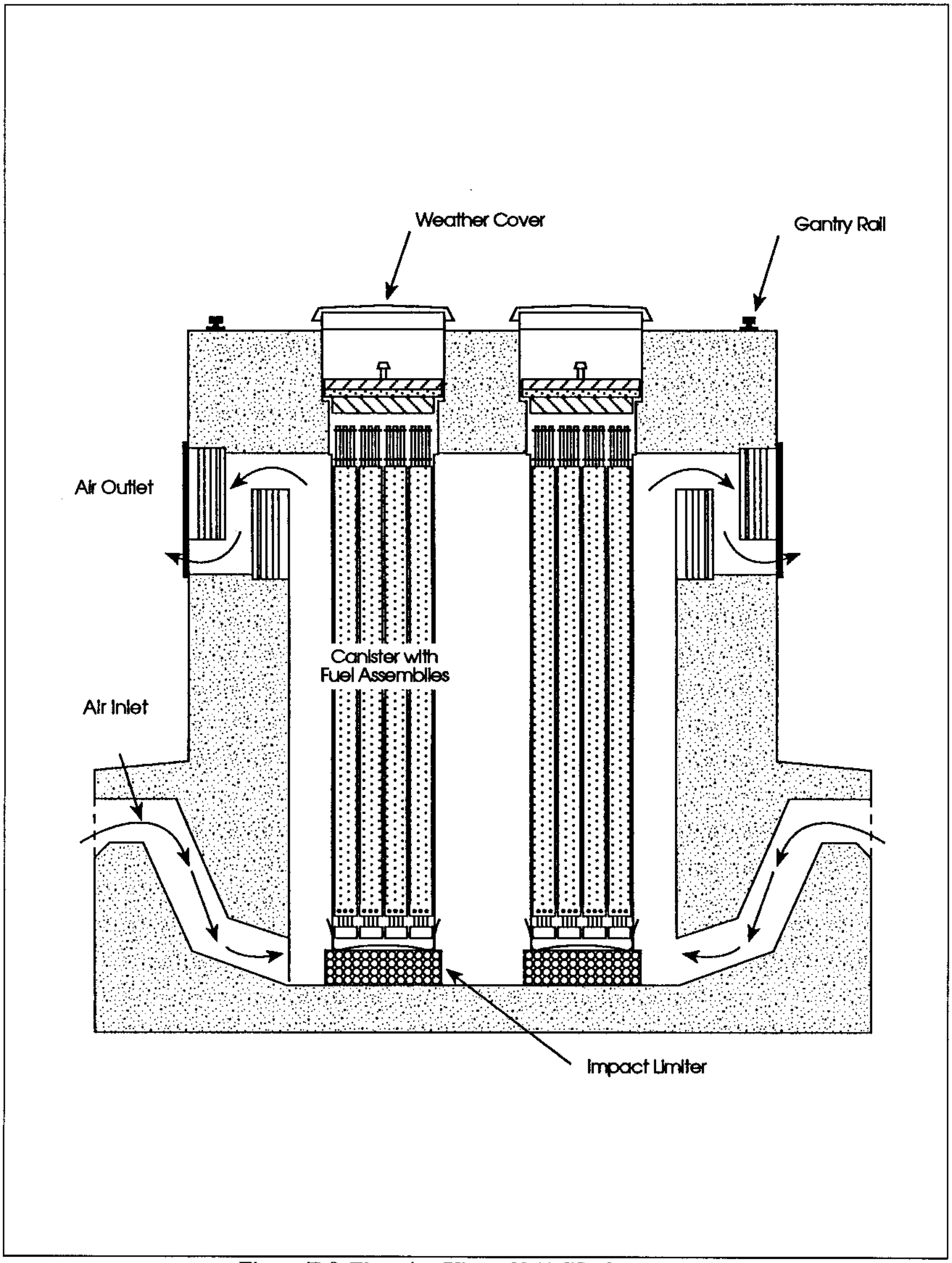


Figure F-8 Elevation View of MACSTOR Module

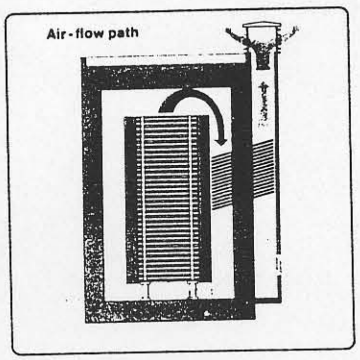
FUELSTOR

FUEL ENCAPSULATION AND LAG STORAGE FACILITY

For at-reactor dry storage of spent fuel

Main technical features:

- Closed-cycle vault resulting in triple barrier system
- Passive cooling by natural convection ensuring low fuel temperatures
- Low initial storage temperature resulting in no restrictions regarding the length of the storage period
- Sealed canister storage
- Horizontal placement of helium-filled canisters in storage vault
- Reinforced concrete construction for safe shielding resulting in radiation levels outside the building below permissible dose of unrestricted area (ALARA)
- Reinforced concrete construction to withstand the effects of natural phenomena and man-induced events such as earthquakes, tornado missiles and aircraft crashes
- Nuclear criticality safety is ensured by the building design as well as spent fuel canister arrangement
- Store is easily expandable by modules
- Low land consumption due to compact spent fuel storage
- Nuclear safeguard provisions



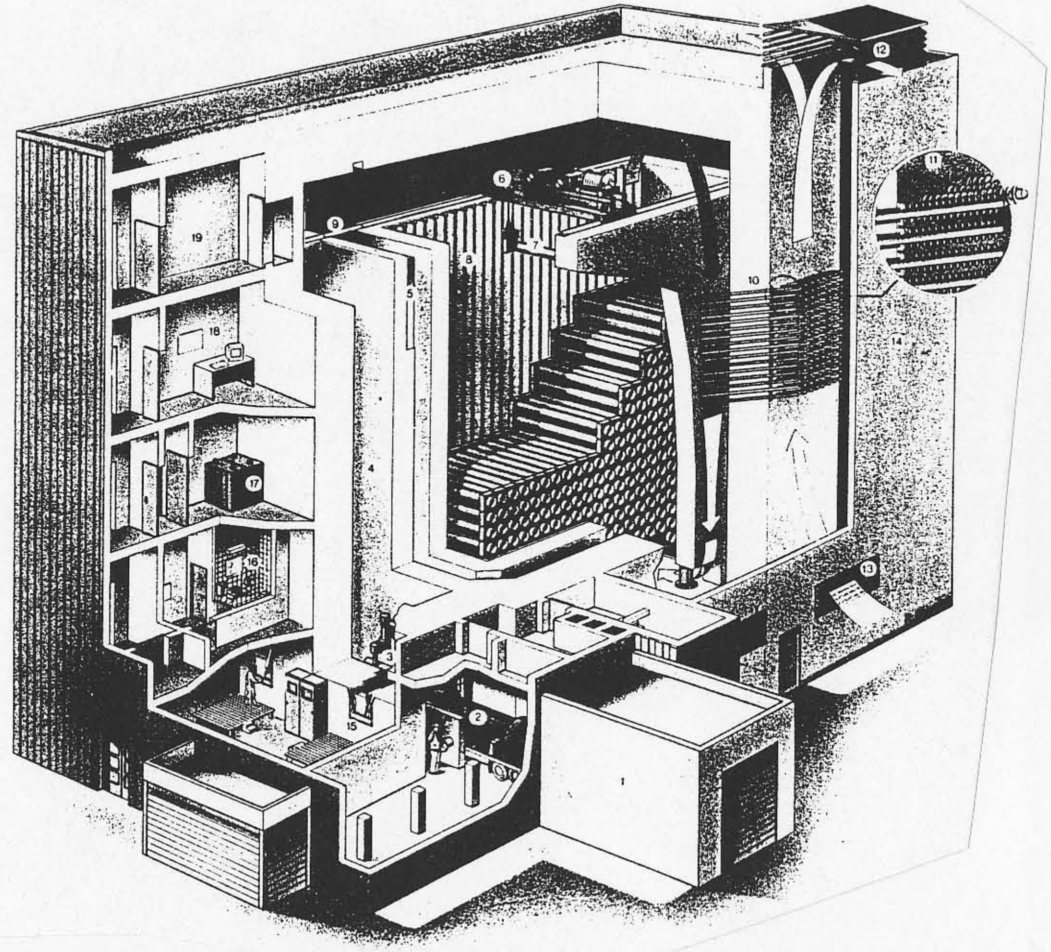


Figure F-9 Fuel Encapsulation and Lag Storage System Cross Section

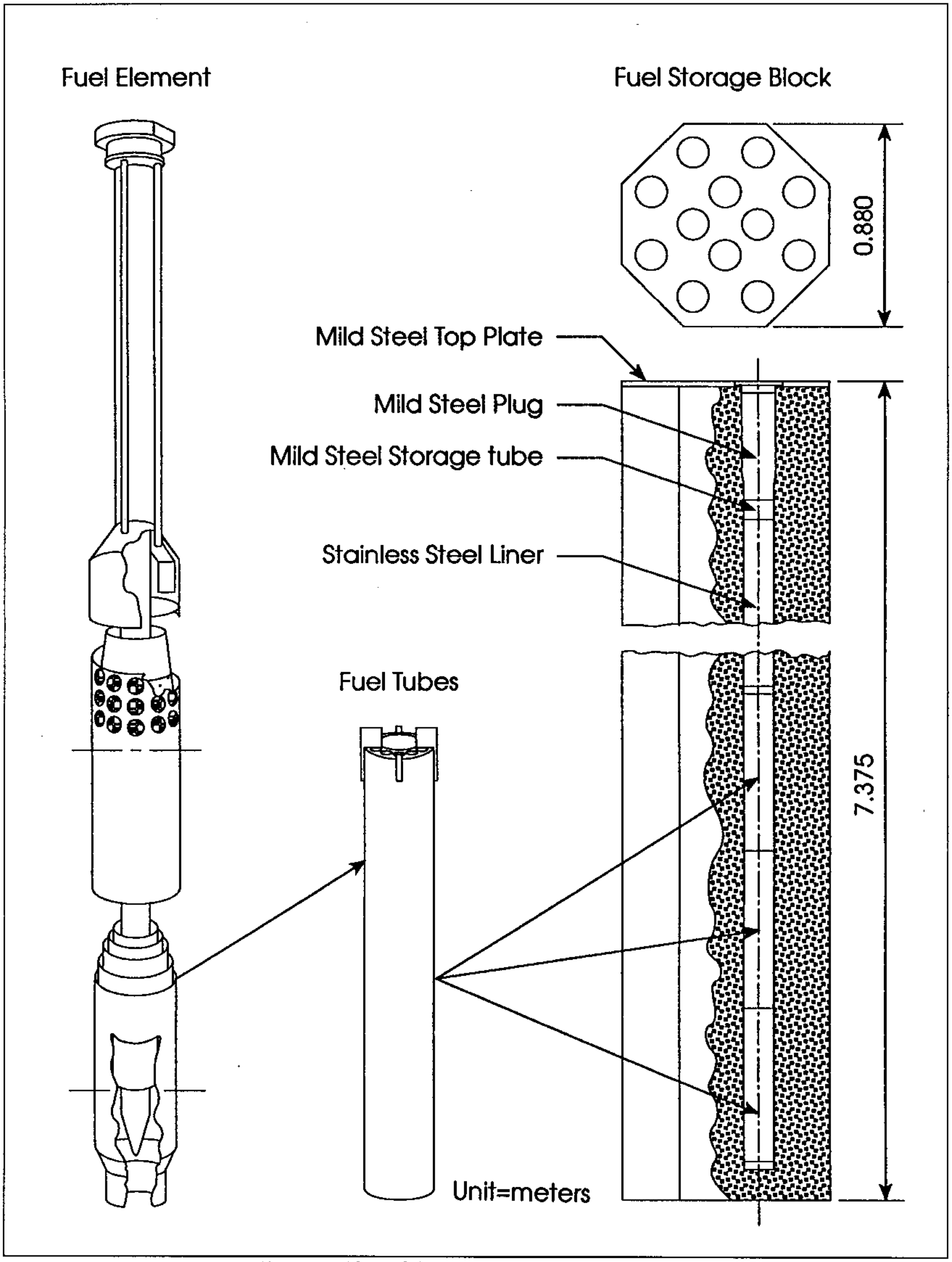


Figure F-10 RISO National Laboratory Design

research reactor spent nuclear fuel. The SILO has been licensed in Canada and is currently undergoing license approval in South Korea, which uses the same regulations as the NRC. A sketch of the SILO is given in Figure F-11.

F.1.1.2.5.5 Dual-Purpose Cask and Canister Systems

Dual-purpose designs must satisfy NRC requirements for both storage and transportation (10 CFR Parts 72 and 71, respectively). It is believed that such dual-purpose designs would reduce incident-free handling of individual spent nuclear fuel assemblies, reduce the volume of low-level radioactive waste that would otherwise be generated from using a single-purpose cask system (one cask for storage with subsequent transfer of individual assemblies to transport casks and disposal packages), and may play a role in reducing overall worker radiation exposures over a single-purpose cask system.

At the present time, there are two dual-purpose casks for light water reactor fuel use: Nuclear Assurance Corporation's Storage/Transport Cask and Vectra's dual-purpose canister system (MP-187). Nuclear Assurance Corporation's Storage/Transport Cask has received NRC approval. The NRC is expected to approve Vectra's MP-187 in the near future.

The VECTRA MP-187 is a derivative of a design approved earlier by the NRC. The MP-187 design includes a stainless steel confinement canister, a horizontal reinforced concrete module for storing the canister, and a special onsite/offsite transportation cask system that may also be used to store the canister in a vertical orientation. This system is currently being evaluated by the NRC for the Rancho Seco nuclear power plant. The applicant also has a variation for the canister design to accommodate canned spent nuclear fuel for damaged spent nuclear fuel assemblies, and which cannot be stored without a second confinement barrier.

DOE had proposed expanding the role of a dual-purpose system to that of a multi-purpose canister-based system (DOE, 1994f; DOE, 1994b; DOE, 1994c). Fuel would be loaded into a canister at the reactor site. The canister could then be placed into unique, specially designed overpacks for storage at the reactor site, transportation to a federal facility, or disposal in a repository. Final NRC approval for use of the multi-purpose canister as a component of the disposal package requires that the multi-purpose canister and its surrounding overpack meet 10 CFR Part 60 requirements. The fact that no site has been chosen yet for a repository adds an element of uncertainty to the third function: disposal. DOE has decided in November 1995 to withdraw its proposal to prepare the EIS for this canister. The Department of the Navy, however, will complete this EIS and will limit its scope to the storage and transport of Navy spent nuclear fuel.

F.1.2 Wet Storage Designs

In addition to the previous examples of dry storage technology, there are several types of wet storage systems currently in use at DOE sites and at commercial nuclear power facilities. These include aboveground pools (lined or unlined), inground pools (lined or unlined), and shutdown reactor vessels. For the purposes of this appendix, a pool refers to a canal or a basin.

Description of Wet Pool Spent Nuclear Fuel Storage Technology

The storage of spent nuclear fuel in pools (i.e., wet storage technology) has been in use for over 40 years (since the early water-cooled reactors began operating). The basic concept underlying wet storage is analogous to the development of light water-cooled nuclear reactors for defense, research, and electric power production purposes.

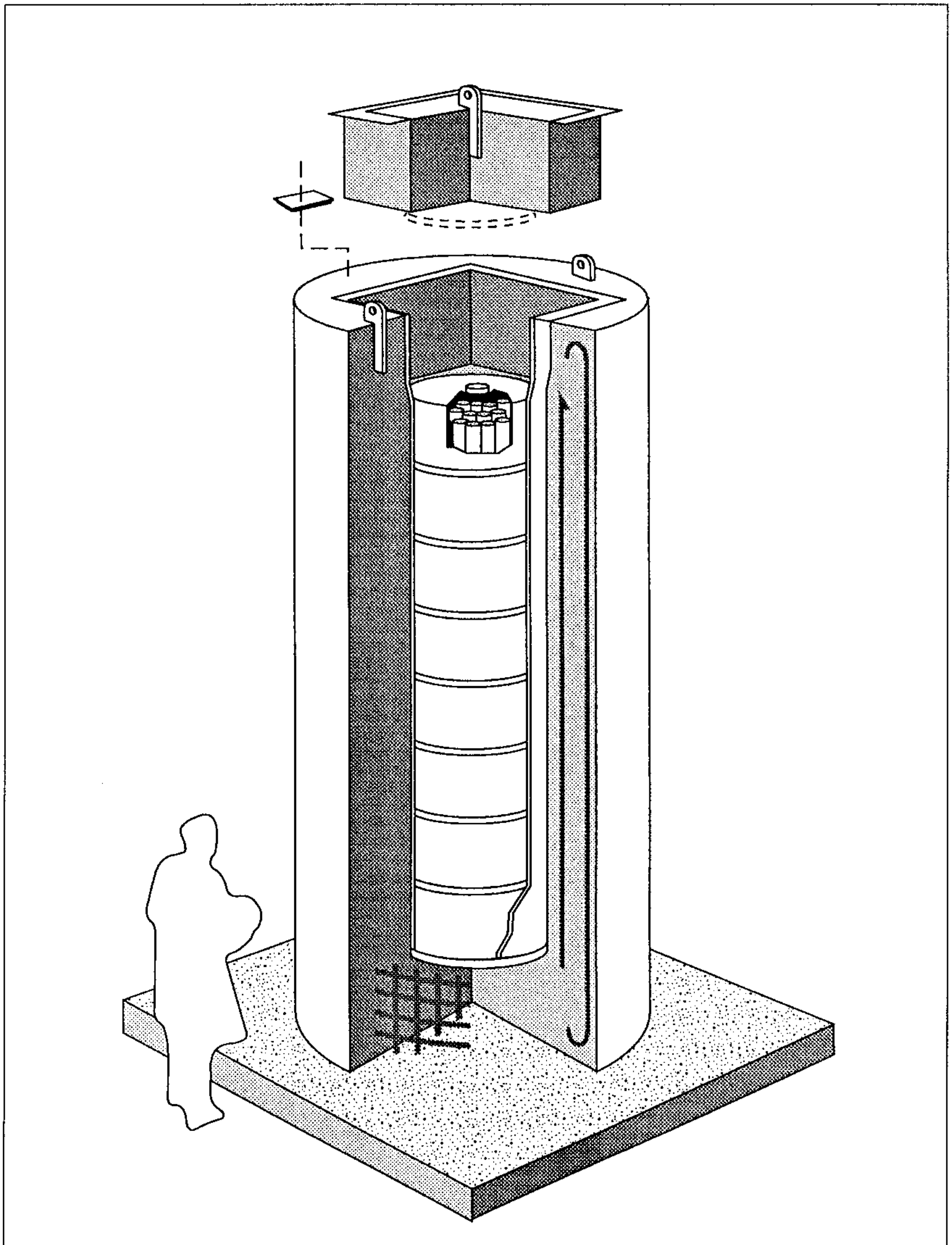


Figure F-11 The Atomic Energy of Canada, Ltd. Concrete SILO

In terms of spent nuclear fuel storage, water offers several distinct advantages, which can be summarized as:

- low cost for shielding and coolant medium,
- visual confirmation of fuel location and ease of handling,
- high heat capacity allowing for a large time period before thermal limits are exceeded,
- multi-purpose shield for both neutrons and gamma rays,
- inherent ability to retain many fission products which could leak from failed spent nuclear fuel, and
- insusceptibility to degradation from spent nuclear fuel radiation.

Water pool storage also has some shortcomings. These are:

- the need to maintain high purity water to prevent corrosion,
- the requirement for active safety systems connected to the water for heat removal, purity control, and water makeup,
- extensive lined and reinforced concrete walls for ensuring no leakage of water under all accident conditions,
- generation of radioactive waste from degraded fuel which is collected by the water purification systems, and
- groundwater monitoring to detect any leakage of radioactive pool water into the environment.

For every water-cooled reactor in the world, the decision has always been made to construct an adjoining or integral spent nuclear fuel storage pool. Currently, over 600 water cooled electric power-, research-, and defense-related reactors are operating in the world, each with its own wet storage pool for spent nuclear fuel (Nuclear Engineering International, 1993). Experience has shown that this technology is safe and effective.

At commercial nuclear power plants, the pool storage is located in a structure adjacent to a containment building that is capable of direct hydraulic connection to the reactor core through a system of canals, gates, and pools. The spent nuclear fuel pool building is designed and built to withstand all the accidents and dynamic loads required of other safety-related structures at nuclear power plants. It has its own crane and fuel handling equipment, and a separate heating, ventilation, and air conditioning system to mitigate radioactive releases to the environment. The nuclear power plant control room includes monitors and controls for the spent nuclear fuel pool. Redundant separate trains of equipment are used to fulfill the requirements of heat removal from the spent nuclear fuel pool water, removal of impurities and radioactive materials from the water, and maintenance of the water level to ensure adequate shielding above the spent nuclear fuel. At commercial power plants, such parameters as water level, water temperature, flow and temperature difference across heat exchangers used to cool the water, water purity, activity levels, and radiation dose rates are all monitored and measured.

All U.S. commercial nuclear power plant pools are stainless steel lined and use racks made of stainless steel to store spent nuclear fuel. Stainless steel is used to line the pool walls and floor to help maintain high water purity by preventing the release of chemicals from unlined concrete and to simplify decontamination at the end of the facility's life. The racks provide support and spacing for each fuel assembly, thus controlling criticality and maintaining fuel structural integrity.

Detailed criticality and thermal-hydraulic analyses are performed to demonstrate to the licensing authorities (the NRC in the United States) that fuel can never become critical, and that the assembly spacing in the racks allows for adequate cooling so as to prevent nucleate and bulk boiling in the pool or on any fuel surfaces. Shielding analyses substantiate the adequacy of the water depth above the fuel in the pool (usually at least 6.1 m or 20 ft), and the thickness of concrete pool walls and piping routing for systems connected to the pool water. This piping may contain pool water that is contaminated with radioisotopes released from spent nuclear fuel in the pool, and must be considered in dose rate evaluations. The shielding analyses provide assurances that the dose rate levels are acceptably low to workers around the spent nuclear fuel pool. Accident analyses are performed to show that the most conservative effects to the public of a postulated release of failed spent nuclear fuel fission products in the pool meet all regulatory dose rate limits.

One difference between nuclear power plant spent nuclear fuel wet storage and that which would be used for foreign research reactor spent nuclear fuel is that at nuclear plants, the pools include soluble boron in the water as a means of controlling criticality. Boron is a powerful neutron absorber, and as such prevents the approach to criticality since neutrons are needed to initiate and maintain a uranium fission chain reaction. Soluble boron would not be used in a wet storage facility for foreign research reactor spent nuclear fuel because it would corrode the aluminum cladding materials present in most foreign research reactor spent nuclear fuel. If neutron-absorbing materials were deemed desirable for foreign research reactor spent nuclear fuel wet storage, they could be incorporated as solid boron-aluminum alloy plates encased in aluminum or stainless steel that are integral to the storage rack design so that boron is physically present between each fuel assembly. This design has been successfully licensed and operated at many commercial nuclear power plants to allow for a higher density or tighter packing of the spent nuclear fuel assemblies.

Figure F-12 illustrates a typical wet pool storage facility for spent nuclear fuel. The operational experience of wet storage facilities is excellent, with no significant accidents or events. In the few instances when fuel was damaged while being moved into or out of the pool, the water mitigated any radiological consequences to workers, the public, or the environment. Some events involved temporary loss of pool water heat-removal systems. The large heat capacity of the water in the pool reduced any increase in fuel temperature so that no harmful effects resulted from such a loss in cooling capacity. These two types of wet storage facility events emphasize the principal benefit of water as a coolant and shielding medium, namely its very large thermal inertia and shielding/radionuclide retention capacity.

There is a long and successful history of safe operation for wet storage of spent nuclear fuel in the commercial power, research, and defense sectors of the nuclear industry. The technology is well known, licensed, and offers extensive operational experience.

The same arguments that apply to dry storage of spent nuclear fuel also apply to wet storage facilities. 10 CFR 72 applies to both dry and wet storage facilities for spent nuclear fuel at commercial licensees. DOE Order 6430.1A (DOE, 1989a) applies to the general design of nuclear facilities on DOE sites, including those for spent nuclear fuel storage. This DOE order references 10 CFR 72 for most of the specific details on spent nuclear fuel storage.

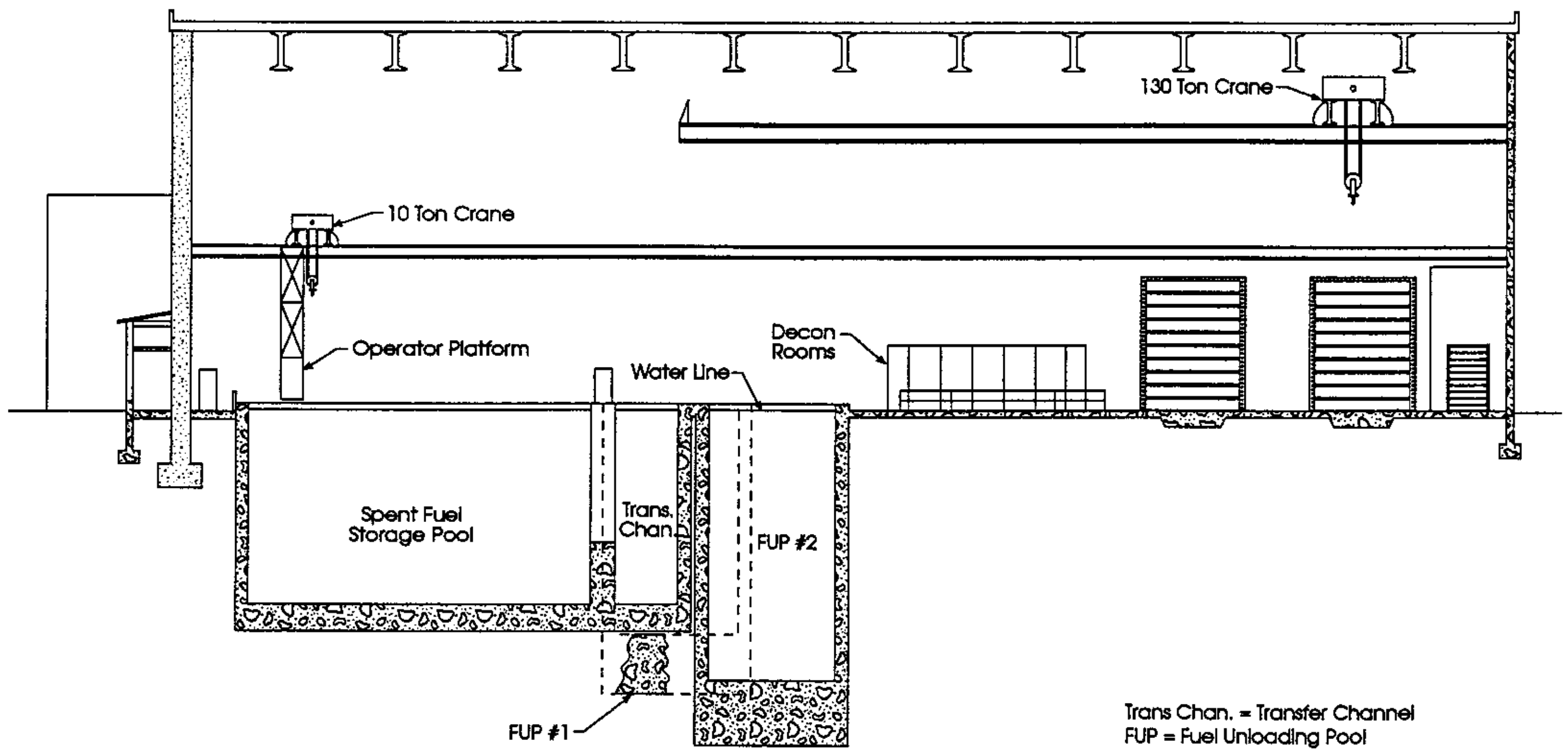


Figure F-12 Typical Wet Pool Storage Facility for Spent Nuclear Fuel

F.1.3 Summary of DOE Spent Nuclear Fuel Locations and Activities

DOE currently has about 2,700 MTHM of spent nuclear fuel in its storage facilities across the DOE complex (DOE, 1994h). Additional generation of about only 100 MTHM is anticipated during the next 40 years. Most of the spent nuclear fuel storage occurs at three sites: Hanford Site (77 percent), Idaho National Engineering Laboratory (10.9 percent), and Savannah River Site (7.3 percent) (Table F-9). Note that the quantities of DOE spent nuclear fuel completely dwarf the expected amount of foreign research reactor spent nuclear fuel (about 19 MTHM) on an MTHM basis (i.e., foreign research reactor spent nuclear fuel is less than 1 percent of the total). However, on a volume basis, foreign research reactor spent nuclear fuel represents about 10 percent of the total and, thus, their storage facilities would be of a significant size. Predominantly wet storage is used at DOE sites, although some limited experience exists with dry storage (e.g., Los Alamos National Laboratory and Idaho National Engineering Laboratory).

Table F-9 DOE Spent Nuclear Fuel Inventory[a, b]

Generator or Storage Site[c]	*Existing (1995)*		*Future Increases (through 2035)*		*Total (2035)*	
	MTHM[d]	*Percent*	*MTHM[d]*	*Percent*	*MTHM[d]*	*Percent*
Hanford Site	2,132.44	80.6	0.00	0.0	2,132.44	77.8
Idaho National Engineering Laboratory[e]	261.23	9.9	12.92	13.5	274.14	10.0
Savannah River Site	206.27	7.8	0.00	0.0	206.27	7.5
Naval Nuclear Propulsion Reactors	0.00[f]	0.0	55.00	57.6	55.0	2.0
Oak Ridge Reservation	0.65	<0.1	1.13	1.2	1.78	<0.1
Other DOE Sites	0.78	<0.1	1.50	1.6	2.28	<0.1
Non-DOE Domestic Research Reactors[g]	2.22	<0.1	3.28	3.4	5.50	0.2
Special-Case Commercial Reactors[h]	42.69	1.6	0	0	42.69	1.6
Foreign Research Reactors[i]	0	0	21.7	22.7	21.70	0.8
Total	**2,646.27**		**95.53**		**2,741.80**	
Percent of 2035 Total	**96.5**		**3.5**		**100.00**	

[a] *Source: DOE, 1995g*

[b] *Numbers may not sum due to rounding.*

[c] *The Nevada Test Site does not currently store or generate spent nuclear fuel and is not expected to generate spent nuclear fuel through 2035. However, in the 2010-2020 timeframe, a repository may open, with annual capacity over 1,000 MTHM.*

[d] *One MTHM equals approximately 2,200 pounds.*

[e] *Sum of fuel located at the Idaho National Engineering Laboratory.*

[f] *Existing inventory of Naval spent nuclear fuel is included in the Idaho National Engineering Laboratory totals (9.95 MTHM).*

[g] *Includes research reactors at commercial, university, and Government facilities.*

[h] *This total is just that stored at non-DOE facilities (Babcock & Wilcox Research Center and Fort St. Vrain). The total inventory of spent nuclear fuel from special-case commercial reactors is 186.41 MTHM. This fuel is also stored at the Idaho National Engineering Laboratory, the Oak Ridge Reservation, the Hanford Site, the Savannah River Site, and the West Valley Demonstration Project.*

[i] *At the Savannah River Site and the Idaho National Engineering Laboratory.*

Wet Storage

DOE spent nuclear fuel pools are in many cases more than 20 years old and were originally unlined, due to simplicity and the relatively short planned duration (3 to 6 months) of spent nuclear fuel storage prior to reprocessing. The spent nuclear fuel storage basins are concrete with 30 to 90 cm (1 to 3 ft) thick walls, and the bottom is usually thicker than the sides. For shielding purposes, the pool maintains a minimum of 3 m (10 ft) of water over the spent nuclear fuel at all times. Thus, total water depth typically ranges between 4.5 to 6.1 m (15 to 20 ft), although some facilities extend to 9.1 m (30 ft). Steel, stainless steel, or aluminum racks are affixed to the bottom of the pool for holding the spent nuclear fuel in a vertical configuration. The spent nuclear fuel basins provide for recirculation and heat removal capabilities, but limited water clarification and purification. Chemicals from the exposed concrete increase pool turbidity and tend to accelerate corrosion phenomena, particularly for aluminum-clad fuels. Some of the DOE spent nuclear fuel wet storage facilities do not meet the present construction requirements.

Wet storage still remains the predominant technology for storing irradiated materials (DOE, 1993b; Taylor et al., 1994). Currently, there are some 29 DOE spent nuclear fuel storage pool facilities in the complex, ranging in age from 10 years to more than 40 years. Facilities built more than 30 years ago were constructed to standards far less rigorous than exist today. Several DOE orders address spent nuclear fuel storage facilities indirectly, while DOE Order 6430.1A specifically sets the design criteria that addresses storage facilities (spent nuclear fuel facilities that are part of a reactor facility are covered by DOE Order 5480.6). Most DOE storage pools were not designed for long-term storage of spent nuclear fuel and targets and have very limited space available for consolidation.

Most of the storage pool surfaces are bare concrete A few are lined with stainless steel, and some are coated with epoxy or vinyl. The unlined pools are more susceptible to leakage and to increased contamination by soluble radionuclides. The unlined bare concrete storage pools do not have effective leak-detection systems to detect and capture potential leaks. To help identify pool leakage, more than 50 percent of DOE storage pools have had groundwater monitoring wells installed.

Severe corrosion of materials within many of the DOE storage pools has occurred. Corrosion has been generally attributed to poor water quality control and material incompatibilities, which has led to pitting and galvanic corrosion of spent nuclear fuel and storage equipment. This could potentially create a problem when the spent nuclear fuel materials have to be moved. In some cases, equipment failure could cause fissile material reconfiguration, which could increase nuclear criticality concerns. As a result of corrosion, release of radionuclides and fissile material to the pools has occurred. Corrosion also creates handling, packaging, inventory control, waste generation, and cleanup problems with the storage pools.

Savannah River Site and DOE (Taylor et al., 1994) consider the following facilities potentially suitable for near-term future wet spent nuclear fuel storage (in some cases with facility upgrades):

- Idaho National Engineering Laboratory
  - Power Burst Facility Canal
  - Idaho Chemical Processing Plant (ICPP)-666 Pool
  - Expended Core Facility
- Savannah River Site
  - 105-K Disassembly Basin

- 105-L Disassembly Basin
- 105-C Disassembly Basin
- 105-P Disassembly Basin
- Receiving Basin for Offsite Fuels (RBOF) Facility (244-H)
- BNFP (acquisition required).

However, only the ICPP-666 pool and the BNFP were found to meet all current standards, and, thus, be considered suitable for long-term storage.

DOE has improved some of its spent nuclear fuel facilities and has plans for additional upgrades (DOE, 1993b; DOE, 1995g). Typical upgrades include:

- installation and operation of water purification equipment, such as demineralizer columns and filters,
- reracking and fuel consolidation to increase fuel storage space, and
- improving seismic resistance (where possible, via additional supports).

These upgrades would extend the life of existing facilities and allow safe storage of spent nuclear fuel until new facilities are constructed or the spent nuclear fuel is chemically separated. In addition, spent nuclear fuel suspect of leaking during this interim period would be removed and canned to extend its safe storage.

Dry Storage

DOE has fewer dry storage facilities, and these range from approximately 1 to 50 years in age. There are many different types and applications of dry storage used throughout the DOE complex. Spent nuclear fuel is sorted in steel structures; lined and unlined concrete hot cells; steel-lined; concrete; below-grade vaults; reprocessing canyon dissolver cells; cans contained in steel wells; and large, above-grade storage casks. Spent nuclear fuel has been characterized and stored in dry configurations within hot cell facilities since the 1950s. Most DOE hot cells were not designed and built for long-term storage of spent nuclear fuel. Their primary mission was to conduct tests and basic research on irradiated fuels resulting in very limited capacity for storage of spent nuclear fuel.

Since the 1970s, spent nuclear fuel has been stored in facilities specifically engineered for longer-term dry storage. Modern dry storage methods in newer facilities provide low corrosion environments within sealed barriers for monitored interim retrievable storage. A few examples of dry storage confinement methods include sealed canisters in wells surrounded by concrete and extensive release protection incorporating High Efficiency Particulate Air-filtered ventilation systems. By using current dry storage technology, dry storage facilities could be engineered to withstand severe natural phenomena hazards, fires, and explosions without damage to the fuel or release of radionuclides. Dry storage technologies can be adapted to store many types of damaged and undamaged DOE-owned spent nuclear fuel.

The application of dry storage technologies generally results in fewer environmental, safety, and health issues as compared with wet storage. However, DOE has limited experience with aluminum-clad, high decay heat fuels in dry storage facilities.

Some quantities of spent nuclear fuel may be in dry storage facilities for much longer than originally anticipated. Over the years, several inground steel-lined storage well barriers have had the potential for severe corrosion, which could result in undetected releases to the environment. This is particularly important, because of the inaccessibility of these facilities for inspection and characterization (e.g., Argonne National Laboratory Radioactive Scrap and Waste Facility).

The Savannah River Site and DOE (Taylor et al., 1994) consider the following facilities suitable for near-term dry storage of spent nuclear fuel.[1]

- Argonne National Laboratory-West
  - Hot Fuel Examination Facility
  - Radioactive Scrap and Waste Facility
- Idaho National Engineering Laboratory
  - Test Area North Test Pad
  - ICPP Irradiated Fuel Storage Facility (IFSF)
  - ICPP-749 (Drywells, Second Generation)
- Savannah River Site
  - 221-H (Extensive modification required)
  - 221-F (Extensive modification required).

However, certain DOE requirements, such as DOE Orders 6430.1A and 5480.6, make it likely that these facilities could not qualify for future, long-term, dry storage of spent nuclear fuel. Excluding the Savannah River Site facilities (because of the extensive required modifications), none of these facilities appear to be very useful for long-term spent nuclear fuel storage. Extensive modifications to the facilities would be required to meet seismic criteria and increase the storage capacity or convert existing facilities (e.g., F- and H-Canyons at the Savannah River Site) into suitable dry storage facilities. However, facilities such as the Hot Fuel Examination Facility and Test Area North appear suitable as possible staging and characterization facilities in a dry cask storage approach, based upon the presented information (Taylor et al., 1994).

F.1.3.1 Savannah River Site

The Savannah River Site occupies an area of approximately 800 km^2 (310 mi^2) in South Carolina, in a generally rural area about 40 km (25 mi) southeast of Augusta, Georgia (DOE, 1995g). The Savannah River forms the southwestern border of the Savannah River Site. The Savannah River Site consists primarily of managed upland forest with some wetland areas, and facilities and railways occupy approximately five percent of the Savannah River Site land area. Figure F-13 presents a map of the Savannah River Site with spent nuclear fuel facilities displayed.

1 *Existing facilities in Nevada were not included in the analysis.*

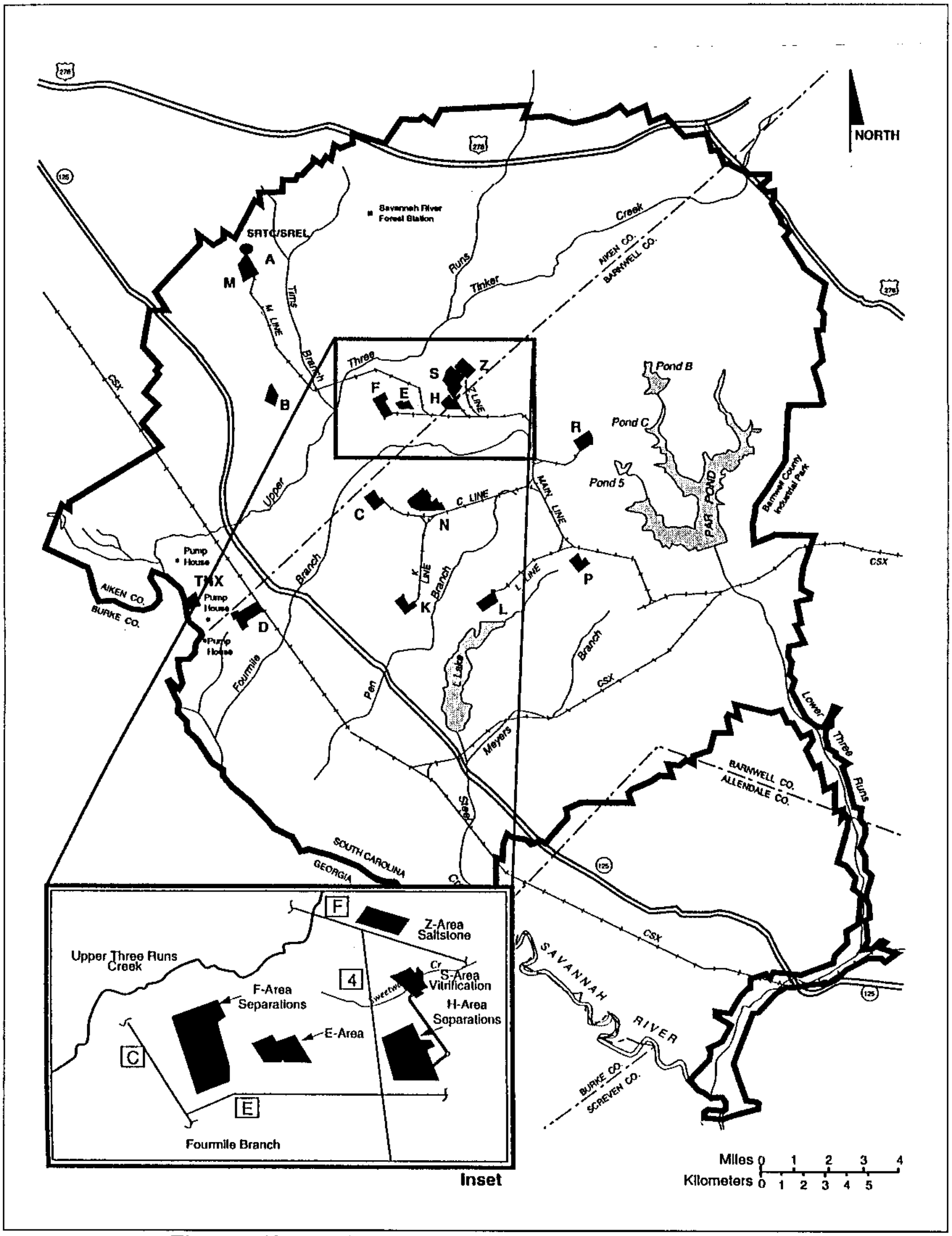


Figure F-13 Location of Principal Savannah River Site Facilities

The primary Savannah River Site facilities were used for the production of nuclear materials. Currently, the production reactor facilities are not operating and are in either shutdown or standby mode. Several large waste management projects are now underway at the site, including the Defense Waste Processing Facility for the vitrification of high-level waste.

F.1.3.1.1 Spent Nuclear Fuel Activities at the Savannah River Site

The Savannah River Site currently stores approximately 201 MTHM of spent nuclear fuel (DOE, 1995g), or approximately 7 percent of the DOE total, including the following:

- 184.4 MTHM of aluminum-based spent nuclear fuel, including plutonium target material,
- 4.6 MTHM of commercial spent nuclear fuel (zircaloy-clad),
- 11.9 MTHM of test and experimental reactor, zircaloy-clad fuel, and
- 5.4 MTHM of test and experimental reactor, stainless steel-clad fuel.

This fuel is stored in several basins onsite. The F- and H-Area Canyons are the processing and separations facilities at the Savannah River Site, and each has a small associated wet storage basin. Three reactor disassembly basins (K, L, and P) contain the reactor fuel and target materials. A fourth reactor disassembly basin (C) currently is the only basin without security upgrades necessary for any storage activities. These basins consist of unlined concrete with inadequate water purification equipment for extended storage of aluminum-clad spent nuclear fuels. These reactor basins were built in the 1950s and were not intended for the long-term storage ("years") of radioactive materials. Furthermore, poor water chemistry has corroded some of the spent nuclear fuel in the K- and L-Reactor disassembly basins, resulting in the release of fissile materials to the pool water. Also, these reactor basins are not seismically qualified and lack modern earthquake resistant features. Ongoing facility upgrades of the L-Reactor disassembly basin are intended to correct the conditions of the basin. Deionization of the basin has lowered the conductivity to acceptable levels for corrosion control. Lower conductivity would greatly reduce the probability of new corrosion and reduce the rate of progression of existing corrosion. The control of the conductivity after the completion of the deionization would be accomplished using the Disassembly Basin Upgrade Project which was initiated to address near term activities and vulnerabilities associated with storing fuel in the L-Reactor disassembly basin. With the upgrades to be completed by mid-1996 (Miller et al., 1995), the L-Reactor basin can be expected to safely store spent nuclear fuel for as long as 10 to 20 years. These upgrades include the following:

- A continuous on-line deionization system to improve water chemistry. The continuous deionization system will lower and control the conductivity levels of the basin thereby minimizing corrosion. The continuous deionizer system also removes ionic radionuclide concentrations, specifically Cesium-137.
- A makeup water deionizer to improve the quality of makeup water supplied to the basin. This action will mitigate any additional load on the continuous deionization system.
- New equipment and systems for alternative packaging and removal of waste.

A Basis for Interim Operation document for the L-Reactor in cold standby conditions was prepared by the Westinghouse Savannah River Company (WSRC, 1995b). The Basis for Interim Operation addressed the effects of process events on the facility worker and the effects of process and natural phenomena hazards events on the public and the environment. The Basis for Interim Operation document concluded that the facility could continue to operate within the safety envelope, identified in the Basis for Interim Operation, without undue risk to the public or the environment.

The RBOF is the other major facility for spent nuclear fuel storage. The RBOF is more suitable than the reactor basins because it is lined (epoxy sides, stainless steel bottom) and has a water purification system. The BNFP, after refurbishing, would be suitable for foreign research reactor spent nuclear fuel storage because it is fully lined with stainless steel, has water purification systems, and has active heat removal systems. Major spent nuclear fuel storage facilities are summarized in Table F-10.

Table F-10 Major Savannah River Site Spent Nuclear Fuel Storage Facilities

Facility	*Characteristics*	*Capacity for Foreign Research Reactor Spent Nuclear Fuel Elements*	*Access*
105-K Disassembly Basin	Basin Dimensions: 46.9 x 65.8 x 5.2m (154'W x 216'L x ~17'D) Basin Water: 13.2 million l (3.5 million gal)	None initially 20,000 after upgrades	Truck/Rail
105-L Disassembly Basin	Basin Dimensions: 46.9 x 65.8 x 5.2m (154'W x 216'L x ~17'D) Basin Water: 13.2 million l (3.5 million gal)	None initially 20,000 after upgrades	Truck/Rail
105-C Disassembly Basin	Basin Dimensions: 39.6 x 58.2 x 5.2m (130'W x 191'L x ~17'D) Basin Water: 13.6 million l (3.6 million gal)	None initially 20,000 after upgrades	Truck/Rail
105-P Disassembly Basin	Basin Dimensions: 55.5 x 68.2 x 5.2m (182'W x 223'L x ~17'D) Basin Water: 18.2 million l (4.8 million gal)	None initially 20,000 after upgrades	Truck/Rail
RBOF (244-H)	Basin 1: 8.2 x 12.1 x 6.7m depth over two-thirds of floor space 8.8m depth over one-third of area Basin 2: 8.2 x 3.9 x 8.8m depth Basin Water: 1.7 million l (450,000 gal)	~1000 initially, plus 1,425 after rearranging[b]	Truck/Rail
BNFP[a]	Several Pools: Main Pool: 14.6 x 14.6 x 9.8m (48'L x 48' x 32'D) Basin Water: 2.1 million l (550,000 gal)	None initially 25,000 after acquisition and reactivation[b]	Truck/Rail

[a] *Discussed in more detail in Section F.1.3.1.3; rail spur not currently active but would be included in reactivation.*

[b] *Difference in capacity between RBOF and BNFP is due to greater pool depth of BNFP and different fuel packing density assumptions for the two facilities.*

F.1.3.1.2 Spent Nuclear Fuel Storage Facilities Available for Foreign Research Reactor Spent Nuclear Fuel at the Savannah River Site

The RBOF is the principal facility applicable for foreign research reactor spent nuclear fuel. This basin has been operating and receiving spent nuclear fuel, including foreign research reactor spent nuclear fuel, since 1964, and is located in H-Area, near the center of the Savannah River Site. The 1,393 m^2 (15,000 ft^2) facility consists of an unloading basin, two storage basins, a repackaging basin, a disassembly basin, and an inspection basin. The basins and their interconnecting canals hold approximately 1,893,000 l (500,000 gal) of water. Spent nuclear fuel elements arrive in lead-lined casks weighing from 22 to 64 metric tons (24 to 70 tons), which a crane lifts from a railroad car or a truck trailer and places in the unloading basin. About 30 percent of the fuels in the RBOF consist of uranium clad in stainless steel or zircaloy, which the Savannah River Site facilities cannot process without modifications. The RBOF is discussed in more detail in Section F.3.

In March 1995, the Savannah River Site estimated that the RBOF has the capacity for approximately an additional 1,000 spent nuclear fuel elements (O'Rear, 1995). However, the Savannah River Site has determined that 1,425 additional spaces can be made available by rearranging fuel in the pools and moving spent nuclear fuel to other storage areas, such as one of the reactor disassembly basins. If empty, the total RBOF capacity would be 6,500 foreign research reactor spent nuclear fuel elements.

F.1.3.1.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Savannah River Site for Foreign Research Reactor Spent Nuclear Fuel

The Savannah River Site is evaluating the use of several new planned or potential facilities for foreign research reactor spent nuclear fuel management. These include:

- a modular dry vault storage building,
- dry cask storage, or
- wet pool storage.

These technologies may require additional support facilities for such functions as: spent nuclear fuel examination, spent nuclear fuel characterization, cask loading and unloading, spent nuclear fuel repackaging, and cask maintenance. The Savannah River Site is also evaluating the use of one or more of the reactor disassembly basins for near-term wet storage of foreign research reactor spent nuclear fuel. These facilities are discussed in more detail in Section F.3.

The Savannah River Site is also evaluating the potential storage of spent nuclear fuel at the BNFP facility. Allied General Nuclear Services constructed a large reprocessing facility for commercial spent nuclear fuel in Barnwell, South Carolina, adjacent to the Savannah River Site (Fields, 1994; Matthews, 1994 and 1991; Taylor et al., 1994; Williams, 1994; WSRC, 1992a-d). This plant was never operated due to a change in Government policy, and was mothballed in the 1980's. The BNFP includes a wet fuel storage basin that is approximately twice the area and potentially has over four times the spent nuclear fuel capacity of the RBOF facility at the Savannah River Site. The wet storage basin is fully lined and seismically qualified and would be capable of storing all of the currently identified foreign research reactor spent nuclear fuel (Jackson, 1994). Facility acquisition, replacement of removed equipment, reactivation, installation of suitable storage racks, and checkout at the facility would be required prior to its use.

Figure F-14 displays the location of the BNFP in relation to the Savannah River Site. This land was originally part of the Savannah River Site. The BNFP site consists of approximately 680 hectares (ha) (1,680 acres).

Allied General Nuclear Services designates the fuel pool area of the plant as the "Fuel Receiving and Storage Stations." Considerable documentation exists for the facility, including the engineering designs, the Environmental Impact Statement (EIS), and the Final Safety Analysis Report submitted to the NRC. The pools and attendant cranes are fully seismically qualified structures. The pool section includes ion exchange systems for pool water purification and a separate radwaste system (solidification may need to be added). The section incorporates capabilities for receipt of either truck or railcarried casks. The main crane is rated at 122 metric tons (135 tons).

The Fuel Receiving and Storage Station facility is shown in Figure F-15 and was designed and constructed to receive, store, and handle spent (irradiated) light water reactor fuel. Spent nuclear fuel assemblies are received in shielded casks by either truck or rail. The assemblies are unloaded underwater and stored underwater to provide cooling and shielding. Stored fuel can be remotely transferred to the adjacent Remote Process Cell and Remote Maintenance and Scrap Cell for mechanical processing. After fuel assemblies are unloaded from the shielded casks, the empty casks are prepared for return shipment.

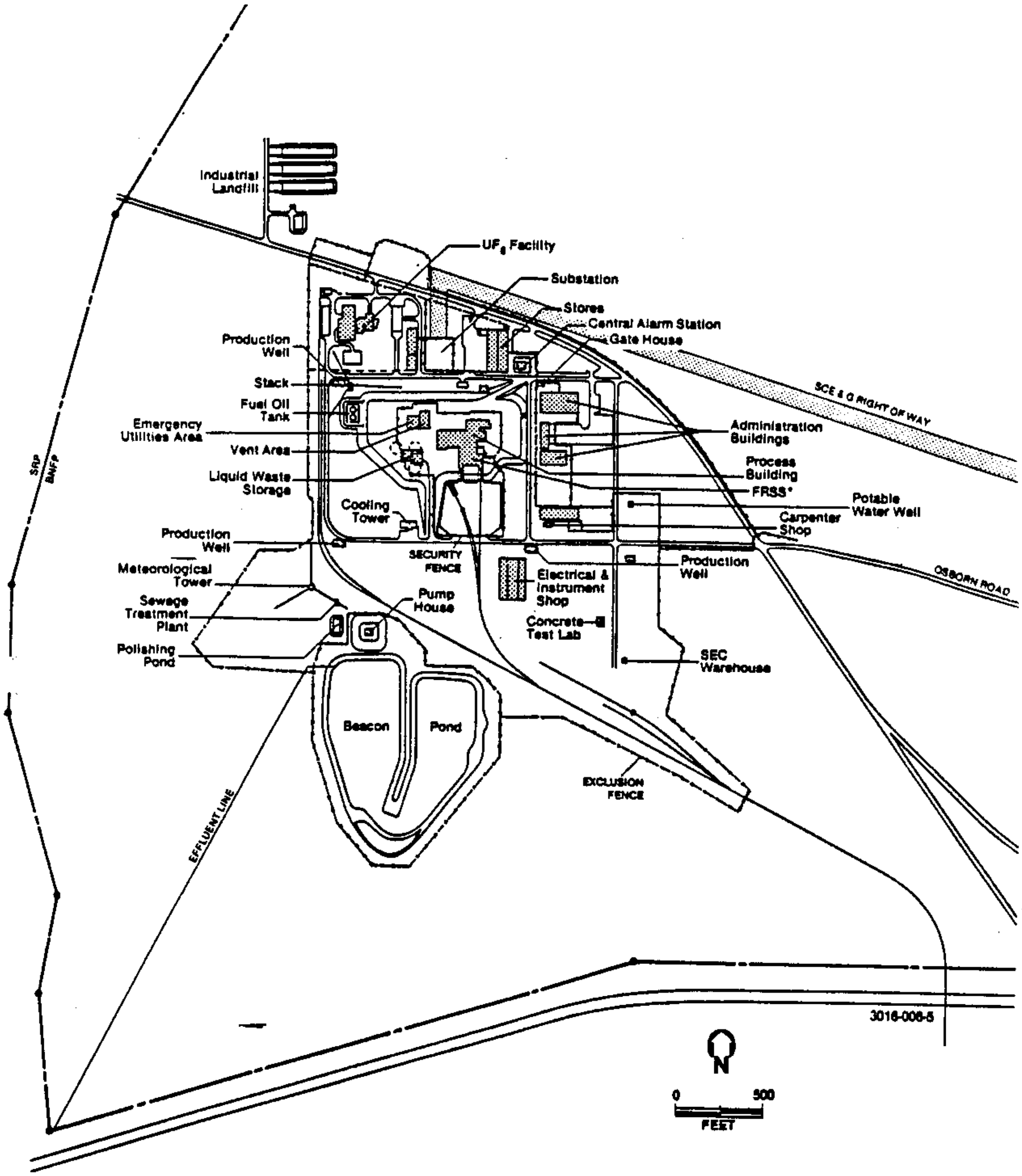


Figure F-14 Plot Plan for the BNFP

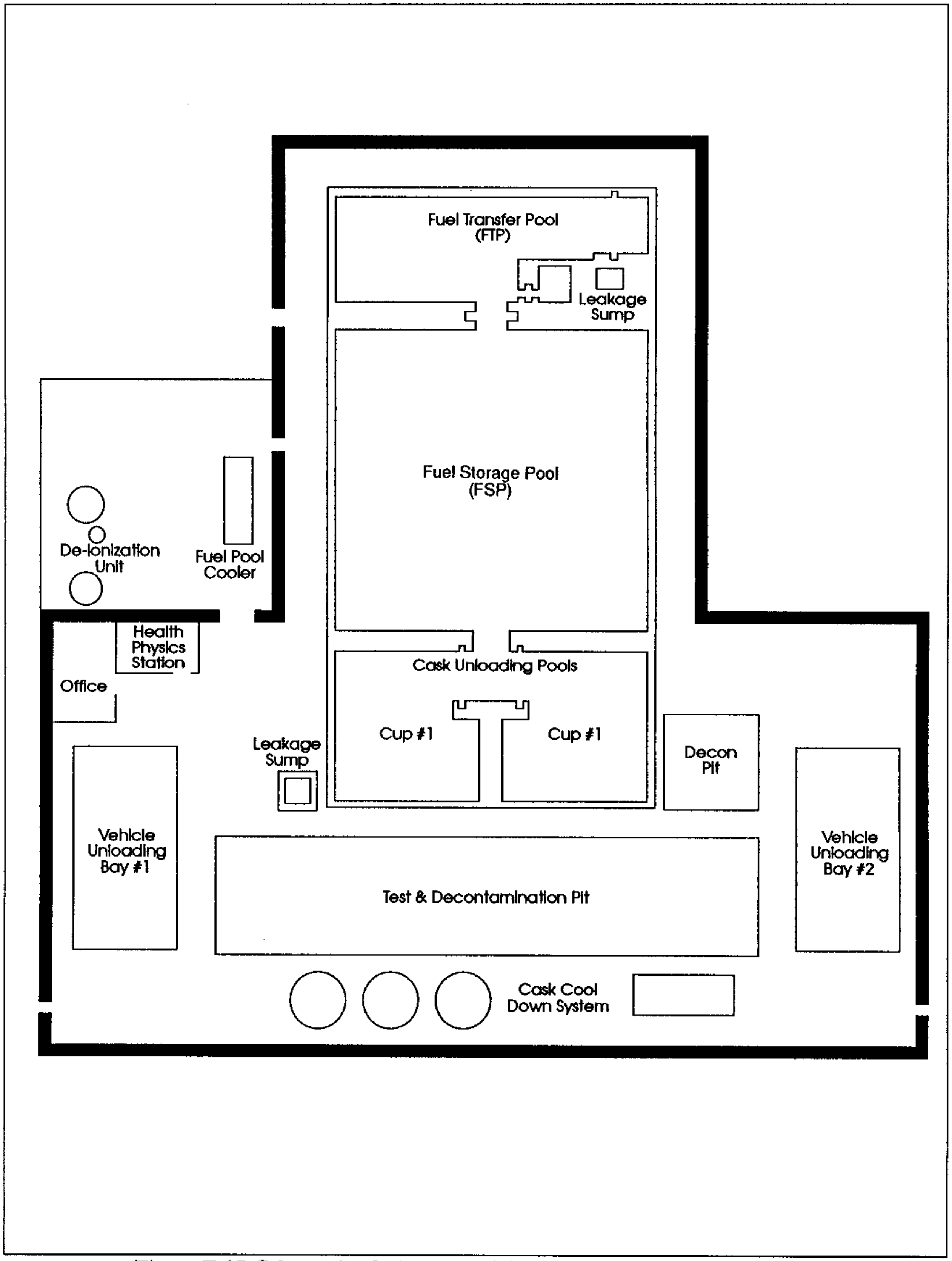


Figure F-15 Schematic of a Fuel Receiving and Storage Station at BNFP

The following areas of the Fuel Receiving and Storage Station are safety class structures:

- pool concrete structure,
- pool and crane column foundations,
- embedments for the fuel storage racks,
- crane rails, rail supports, and restrainer bars which retain the cranes on their rails and prevent their falling into the pools,
- cask barrier beams and embedments,
- energy absorbing pads in the Cask Unloading Pools,
- emergency water supply line, and
- Fuel Receiving and Storage Station walls to the 7.6 m (25 ft) level above grade. Clean spent nuclear fuel casks are moved to the Fuel Receiving and Storage Station water pool area. This area is divided into six pools consisting of two Cask Unloading Pools, one Fuel Storage Pool, one Failed Fuel Pool, one Fuel Transfer Pool, and an examination cell/pool.

Water shielding of 3.7 m (12 ft) is provided in the Fuel Receiving and Storage Station pools. This limits surface dose rates to a calculated 0.08 mrem/hr, assuming design basis Light Water Reactor fuel, and permitting at least 40 hours per week working time for an operator. Handling systems are designed with special limit switches and mechanical stops to prevent raising fuel higher than the design depth of the shielding water.

The water in the five pools of the Fuel Receiving and Storage Station is channeled and treated to promote maximum clarity, to control temperature, and to minimize corrosion and radioactivity. This is accomplished by continuous filtration through 95 percent efficient 5 micron pore size filter elements, cooling in heat exchangers to hold the pool water temperature below 41°C (105°F), and demineralization.

Demineralizing water treatment is designed to maintain radioactivity levels below 0.0005 μCi/ml. Pool water is pumped from the Fuel Storage Pool at 7,570 l/min (2,000 gal/min), directed through the heat exchangers, and returned to the Fuel Storage Pool. A second stream is pumped at 1,135 l/min (300 gal/min) from a pool and is filtered. After filtration, one-half of this stream is treated by ion exchange. The combined filtered and purified solution is then returned to the Fuel Storage Pool. The pool piping system is arranged so that the cleanup stream can be removed from or returned to any of the pool areas, permitting cleanup of contaminated water.

The cooling system is designed to remove heat at a rate of 4,000 kilowatts (14 million Btu/hr). The cooling capacity can be increased by expanding the capacity of the heat exchanger system in the Fuel Receiving and Storage Station. The estimated life for the structure is 50 years (Fields, 1994; Matthews, 1994; Taylor et al., 1994).

The Fuel Receiving and Storage Station has the following six pools:

- two cask pools, each 18.3 m (60 ft) deep,
- failed fuel pool (for degraded fuel),
- fuel transfer pool, 18.3 m (60 ft) deep,

- examination cell/pool, and
- main pool, 14.6 m x 14.6 m x 9.8 m deep (48 ft by 48 ft by 32 ft deep).

All of the pools are lined with stainless steel and are designed to maintain a minimum of 3.7 m (12 ft) of water above the fuel for shielding. The pools include detectors and flow channels for managing potential leaks. The original capacity of the main pool was 400 MTHM. Various analyses have been performed to increase this capacity to the 1,200 to 2,000 MTHM range with reracking and other arrangements. It has been estimated that maximum wet storage corresponds to approximately 5,200 Pressurized Water Reactor assemblies (Taylor et al., 1994). For foreign research reactor spent nuclear fuel, this would correspond to over 25,000 elements; and, thus, as noted previously, the BNFP could accommodate all of the fuel.

The environmental impacts of spent nuclear fuel storage at the BNFP have also been analyzed for between 360 and 5,000 MTHM of commercial fuel (Taylor et al., 1994). The results were:

- Dose commitments to the 80 km (50 mi) population were estimated to be 0.067 person-rem and 0.071 person-rem for 15- and 25-year storage periods, respectively.
- The worst accident would result in a dose commitment of 1 mrem total body, 6 mrem thyroid, and 100 mrem skin to an exposed individual located at the eastern boundary of the site.

These analyses were based upon commercial spent nuclear fuel, but should bound the consequences of foreign research reactor spent nuclear fuel storage at the BNFP. Potential impacts are discussed in more detail in Section F.4.

The BNFP site consists of some 680 ha (1,680 acres), bounded on three sides by the Savannah River Site. Preliminary walkthroughs and analyses by the Savannah River Site indicate the facility is in good condition, and principally needs a main transformer for power supply. The Savannah River Site has estimated a cost of $50 million (Matthews, 1991; WSRC, 1992a-d). Actual acquisition and reactivation costs are claimed to be as low as $25 million (Matthews, 1994; WSRC, 1992a-d). This facility, however, would not be available immediately to receive the foreign research reactor spent nuclear fuel.

Figure F-16 displays the foreign research reactor spent nuclear fuel storage capacity versus time for the Savannah River Site. Clearly, the Savannah River Site can accommodate foreign research reactor spent nuclear fuel at existing facilities supplemented by dry storage, modified reactor disassembly basins, or the potential use of the BNFP. The reactor basins could be used to store the non-aluminum-based spent nuclear fuel currently in the RBOF because the poorer water quality in the basins would not cause additional corrosion for this other fuel that is not aluminum based. Recent improvements in reactor basin water chemistry control have resulted in a substantial decrease in the potential for corrosion of aluminum-clad spent nuclear fuels.

F.1.3.2 Idaho National Engineering Laboratory

The Idaho National Engineering Laboratory has several reactors and critical assemblies operating and also possesses several reactors that are either in standby or shutdown and awaiting decommissioning. From 1953 until 1992, the Idaho National Engineering Laboratory was responsible for processing and recovering highly-enriched uranium (HEU) from naval reactors. The Idaho National Engineering Laboratory discontinued processing spent nuclear fuel in 1992. Consequently, the Idaho National Engineering Laboratory has spent nuclear fuel facilities, spent nuclear fuel in storage, and spent nuclear fuel from

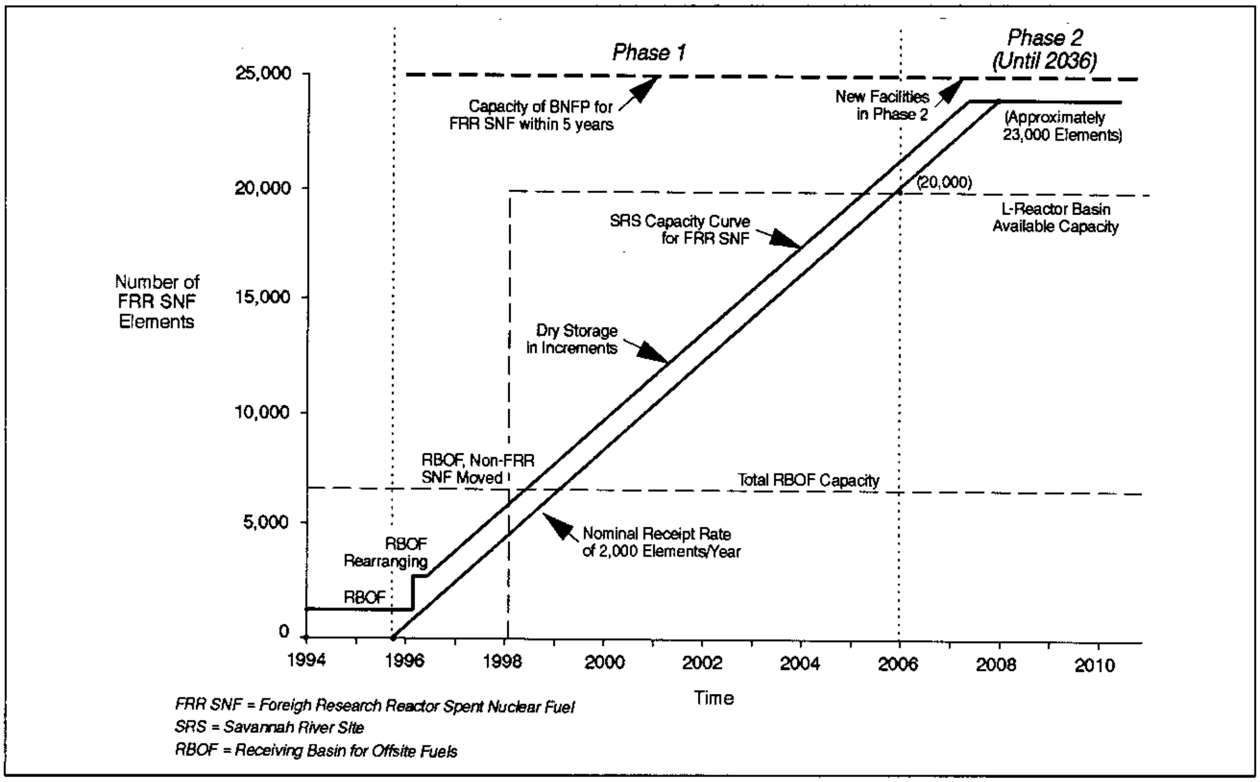


Figure F-16 Foreign Research Reactor Spent Nuclear Fuel Storage at the Savannah River Site

current operations. The Idaho National Engineering Laboratory site map with spent nuclear fuel facilities is shown in Figure F-17.

F.1.3.2.1 Spent Nuclear Fuel Activities at the Idaho National Engineering Laboratory

Six major facility areas at the Idaho National Engineering Laboratory store spent nuclear fuel:

- ICPP,
- Test Area North,
- Power Burst Facility,
- Test Reactor Area,
- Argonne National Laboratory-West, and
- Naval Reactors Facility.

A description of each major facility area and its spent nuclear fuel storage activities is presented below.

F.1.3.2.2 Spent Nuclear Fuel Storage Facilities at the Idaho National Engineering Laboratory

Spent nuclear fuel at the Idaho National Engineering Laboratory is sorted in a variety of dry and wet configurations. The total amount of spent nuclear fuel at the Idaho National Engineering Laboratory

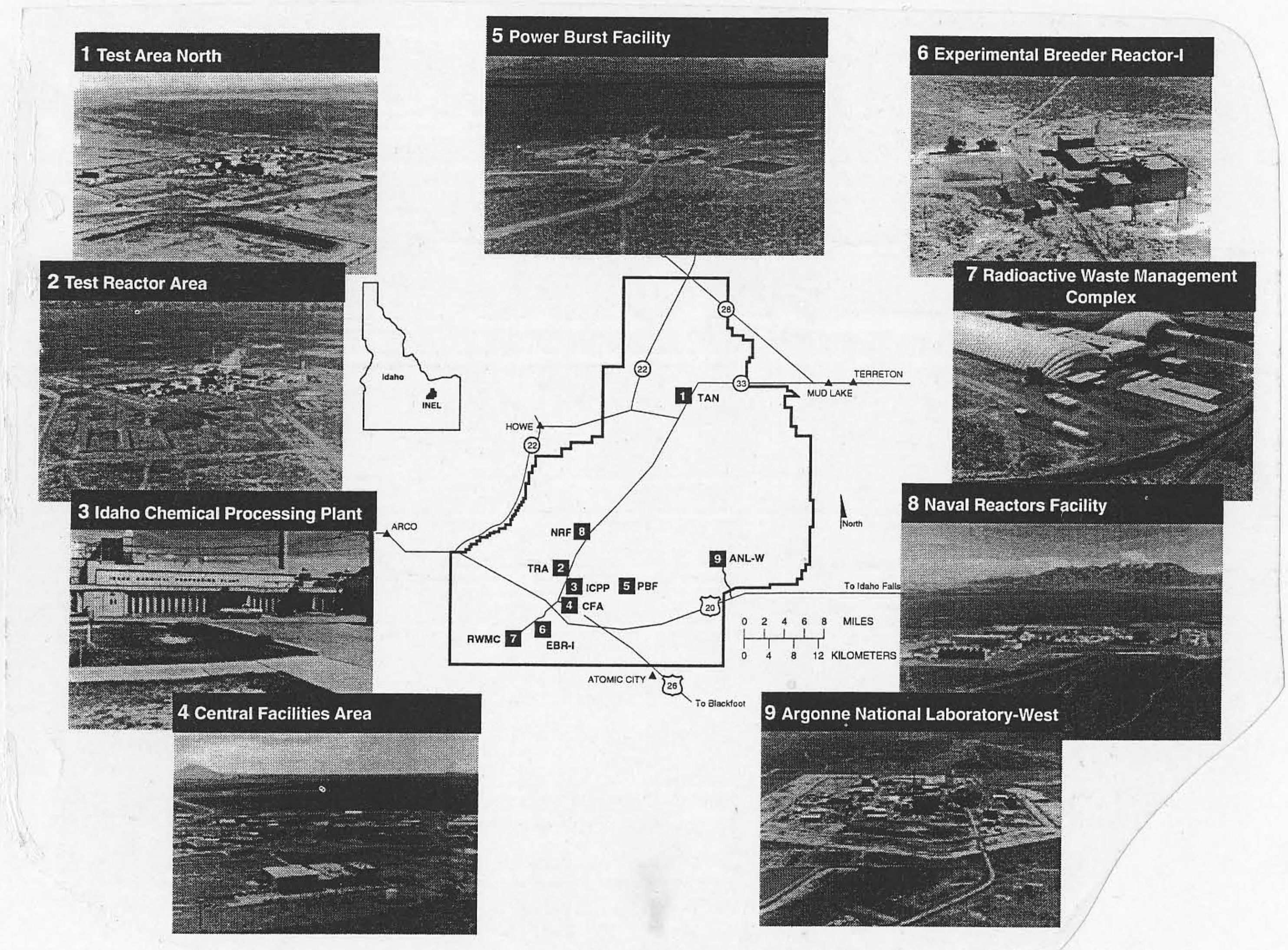


Figure F-17 Location of Spent Nuclear Fuel Storage and Handling Facilities at the Idaho National Engineering Laboratory

accounts for about 10 percent (by weight of heavy metal) of the spent nuclear fuel in the DOE complex (DOE, 1995g).

Table F-11 lists the primary spent nuclear fuel storage facilities, including the type of storage configuration, capacity for foreign research reactor spent nuclear fuel receipts, and accessibility. The number, variety, and location of the wet and dry configurations currently in use at the Idaho National Engineering Laboratory are largely the result of the different purposes for the facilities (e.g., at-reactor storage, storage research and development, reprocessing, and fuel research and development). The condition of the spent nuclear fuel in storage is generally good, with the notable exception of minor amounts of fuel in the Underwater Fuel Storage Facility at the ICPP-603.

The ICPP has received spent nuclear fuel from many onsite and offsite reactors (including foreign research reactor spent nuclear fuel) for reprocessing. Reprocessing for recovery of HEU materials was ceased in 1992. The ICPP now has the mission of managing its current spent nuclear fuel inventory and assigned new spent nuclear fuel receipts, development of technologies in support of dispositioning the spent nuclear fuel, and eventually packaging the material for shipment to a repository. The ICPP stores virtually all types of spent nuclear fuel except production reactor fuel (i.e., fuel from the Hanford Site and the Savannah River Site production reactors). It stores nonproduction reactor aluminum, stainless steel, zirconium, and graphite-clad spent nuclear fuel and uses both wet and dry storage configurations. The ICPP facilities have experience and some capacity for foreign research reactor spent nuclear fuel storage. These are discussed in more detail in Section F.3.

The Test Area North has been a reactor testing facility and has received significant amounts of spent nuclear fuel for examination and testing purposes. This includes the commercial dry storage cask demonstration program and the Three Mile Island debris examination program. It has a very large hot cell and an adjacent underwater storage pool to support the testing programs. It also has a large hot shop where large pieces of equipment, such as transportation casks, have been reconfigured or maintained. At the current time, the Test Area North hot cell and pool have no future mission, but may be used by the U.S. Navy. If Test Area North is not used by the Navy, then the Test Area North hot cell and pool may have significant capacity for receipt of foreign research reactor spent nuclear fuel and for placing it into temporary underwater storage or dry storage casks.

Other storage areas such as the Power Burst Facility reactor canal and the MTR storage pool have limited storage capacities for receipt or storage of foreign research reactor spent nuclear fuel.

The Argonne National Laboratory-West facilities supported the Experimental Breeder Reactor program and also contain the Transient Reactor Test Facility, the Zero Power Physics Reactor, and the Neutron Radiography Reactor. Spent nuclear fuel storage facilities include an at-reactor molten sodium storage pool, in-process lag storage in the Hot Fuel Examination Facility and dry underground SILOs for spent fuel and wastes pending disposition. The Hot Fuel Examination Facility would be suitable for foreign research reactor spent nuclear fuel examination activities.

The Naval Reactors Facility is also located at the Idaho National Engineering Laboratory, but is not included in Table F-11 because of its sole purpose to support the Naval ship propulsion program. The Naval Reactors Facility includes the Expended Core Facility, which receives and examines Naval spent nuclear fuel to support fuel development and performance analyses. In addition, the Expended Core Facility removes structural support material from the Naval spent nuclear fuel before transfer of the fuel portion to the ICPP for reprocessing or interim storage.

Table F-11 Description of Existing Spent Nuclear Fuel Facilities at the Idaho National Engineering Laboratory

Facility	Description	Capacity for Foreign Research Reactor Spent Nuclear Fuel	Access
ICPP-666 Underwater Fuel Storage Area	Water Storage Facility with 6 lined storage basins 9.4 m x 14.2 m by 9.4 m or 12.5 m deep (31 ft x 46.5 ft x 31 ft or 41 ft deep)	Temporary storage after reracking for 8,400 elements	Shipment by truck. Rail shipments to a site receiving area 8 km (5 mi) away.
ICPP-603 Underwater Fuel Storage Area	Water Storage Facility with three basins of varying sizes, no sealant or liner	Not Available - facility is being shut down	Shipment by truck.
ICPP-603 Irradiated Fuel Storage Facilities	Dry Storage Facility with remote unloading area and vault storage with 636 0.5 x 3.4 m L (18 in x 11 ft long) containers	200 containers available for storage of 9,000 foreign research reactor elements	Shipment by truck. Rail shipments to a site receiving area 8 km (5 mi) away.
ICPP-749 Underground Fuel Storage Area	Dry Storage Facility with 218 underground SILOs	Approximately 60 SILOs available following renovation of first generation SILOs; capacity for 3,600 elements after fiscal year 1998	Requires receipt into ICPP-666 or ICPP-603 IFSF and packaging and conditioning for dry storage.
Test Area North-607 Pool and Hot Cell	Water Storage Facility with adjacent remote hot cell	Approximately 56 m^2 (600 ft^2) of basin 7.3 m (24 ft) deep. Capacity for 4,000 elements after new rack installation.	Additional storage space available in hot cell. Shipment by truck, cask unloading in hot cell.
Test Reactor Area-620 Power Burst Facility	Small water storage pool adjacent to Power Burst Facility reactor	Minimal space available	Shipment by truck. Crane capacity inadequate for foreign research reactor casks.
Test Area North-607 Cask Storage Pad	Five commercial fuel storage casks on concrete pad	Easily expandable for more cask storage as necessary for foreign research reactor spent nuclear fuel shipments	Shipment by truck to hot cell where foreign research reactor spent nuclear fuel can be transferred to storage casks and moved to storage pad.
Test Reactor Area-603 MTR Pool	Water Storage Pool in basement of the MTR	Minimal space available	Shipment by truck. Crane capacity and access inadequate for foreign research reactor casks.
Test Reactor Area-660 ARMF and CFRMF Reactors and Canal	Swimming pool reactors with connecting canal	Minimal space available	Shipment by truck. Crane capacity and access inadequate for foreign research reactor casks.
Argonne National Laboratory-West Hot Fuel Examination Facility	Large two room hot cell facility for fuel examinations with argon atmosphere	Minimal space available without extensive removal of examination equipment	Shipment by truck access to hot cell limited to special designed small transfer cask.
Argonne National Laboratory-West Radioactive Scrap and Waste Facility	1,200 vertical steel-lined underground dry storage wells	About 500 wells are not used	Access requires transfer through Hot Fuel Examination Facility.

F.1.3.2.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Idaho National Engineering Laboratory for Foreign Research Reactor Spent Nuclear Fuel

The main focus of near-term activities is the accurate quantification and characterization of DOE-owned spent nuclear fuel, identification of spent nuclear fuel management facilities and their conditions, identification of safe interim storage for existing and new spent nuclear fuel, and identification of technologies and requirements to place DOE spent nuclear fuel in safe interim storage. Long-term activities include the development of final waste acceptance criteria requirements and stabilization technologies for alternate fuel disposition, construction of facilities to stabilize fuel to meet waste disposal requirements, processing of the fuel to a final waste form, and transportation of the waste form for disposition (discussed in more detail in Section F.3). As shown in Figure F-18, the Idaho National Engineering Laboratory has sufficient capacity for foreign research reactor spent nuclear fuel if the existing facilities are supplemented by dry casks.

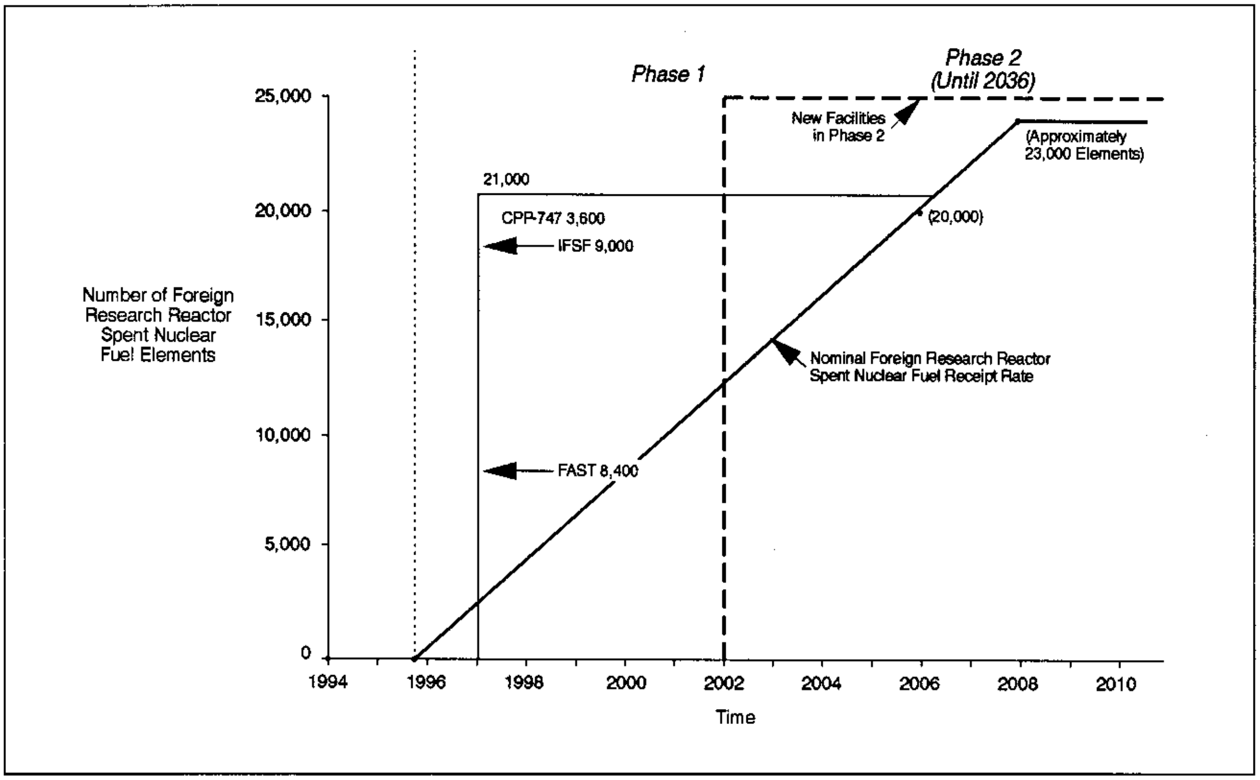


Figure F-18 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Idaho National Engineering Laboratory

F.1.3.3 Hanford Site

The Hanford Site lies within the semi-arid Pasco Basin of the Columbia Plateau in southeastern Washington State (DOE, 1995g). The Hanford Site occupies an area of around 1,450 km^2 (560 mi^2) north of the confluence of the Yakima and Columbia Rivers. Only about six percent of the site has been disturbed in the process of special nuclear materials production for national defense reprocessing and used for DOE purposes, such as nuclear materials production, processing, research and development, and waste management. The Hanford Site facilities include nine shutdown production reactors and several smaller research reactors. Several processing and product finishing facilities are located on the site, but are not currently operating and will not likely operate in the future. Currently, the principal mission of the site is environmental management and includes:

- decontamination and decommissioning of surplus facilities,
- environmental restoration of over 1,500 waste management units and 4 groundwater contamination plumes,

- waste management, including new processing facilities and retrievable disposal, and
- research and development into energy, environmental, and waste management technologies.

A Tri-Party Agreement between DOE, the U.S. Environmental Protection Agency, and the State of Washington provides milestones and guidance for these activities at the Hanford Site. Current schedules use a 2030 date for the completion of most of the restoration activities at the site. A map of the Hanford Site that shows spent nuclear fuel facilities is presented in Figure F-19. Existing spent nuclear fuel facilities are listed in Table F-12.

F.1.3.3.1 Spent Nuclear Fuel Activities at the Hanford Site

The following spent nuclear fuel types and their associated facilities are at the Hanford Site:

- *N Reactor Spent Nuclear Fuel:* This is zircaloy-clad, metallic uranium fuel stored in water in the 105-KE and 105-KW Basins (1,146 and 954 MTHM, respectively), and exposed to air in the plutonium-uranium extraction dissolver cells A, B, and C (0.3 MTHM).
- *Single-Pass Reactor Spent Nuclear Fuel:* This is aluminum-clad, metallic uranium fuel stored in water in the 105-KE and 105-KW Basins (0.4 and 0.1 MTHM, respectively), and stored in water in the plutonium-uranium extraction basin (approximately 2.9 MTHM).
- *Fast Flux Test Facility Spent Nuclear Fuel:* This consists of stainless steel-clad fuel stored in liquid sodium at the Fast Flux Test Facility, comprised mainly of a uranium/plutonium oxide fuel, but with some carbide, metallic, and nitride fuel elements (in all, fuel from 329 assemblies of spent nuclear fuel).
- *Shippingport Core II Spent Nuclear Fuel:* These assemblies are zircaloy-clad uranium dioxide fuel, and are stored in the T-Plant Canyon, Pool Cell 4.
- *Miscellaneous Commercial and Experimental Spent Nuclear Fuel:* This includes primarily zircaloy-clad uranium dioxide fuel stored in air, but does include some Test, Research, Isotope, General Atomic (TRIGA) reactor hydride spent nuclear fuel stored in water and aluminum-clad, uranium-aluminum metallic fuel stored in air. These are principally stored in the 300-Area at Hanford Site.

F.1.3.3.2 Spent Nuclear Fuel Storage Facilities at the Hanford Site

The Hanford Site spent nuclear fuel storage facilities are principally based upon wet methods. Table F-12 provides a brief summary of these facilities. The age, condition, available capacity of these facilities, and the Tri-Party Agreement milestones generally prevent the use of the existing facilities for storage of foreign research reactor spent nuclear fuel. It is extremely unlikely that significant processing activities on spent nuclear fuel will occur in the near future, and thus, new facilities would be required for foreign research reactor spent nuclear fuel management at the Hanford Site.

Two spent nuclear fuel EIS documents address the environmental impacts from spent nuclear fuel management at the Hanford Site. The first is the DOE Programmatic Spent Nuclear Fuel Management and Idaho National Engineering Laboratory Environmental Restoration and Waste Management Programs Final EIS (SNF&INEL Final EIS) (DOE, 1995g), the Record of Decision of which was issued on May 30, 1995, that in general specifies spent nuclear fuel management throughout DOE; and in particular

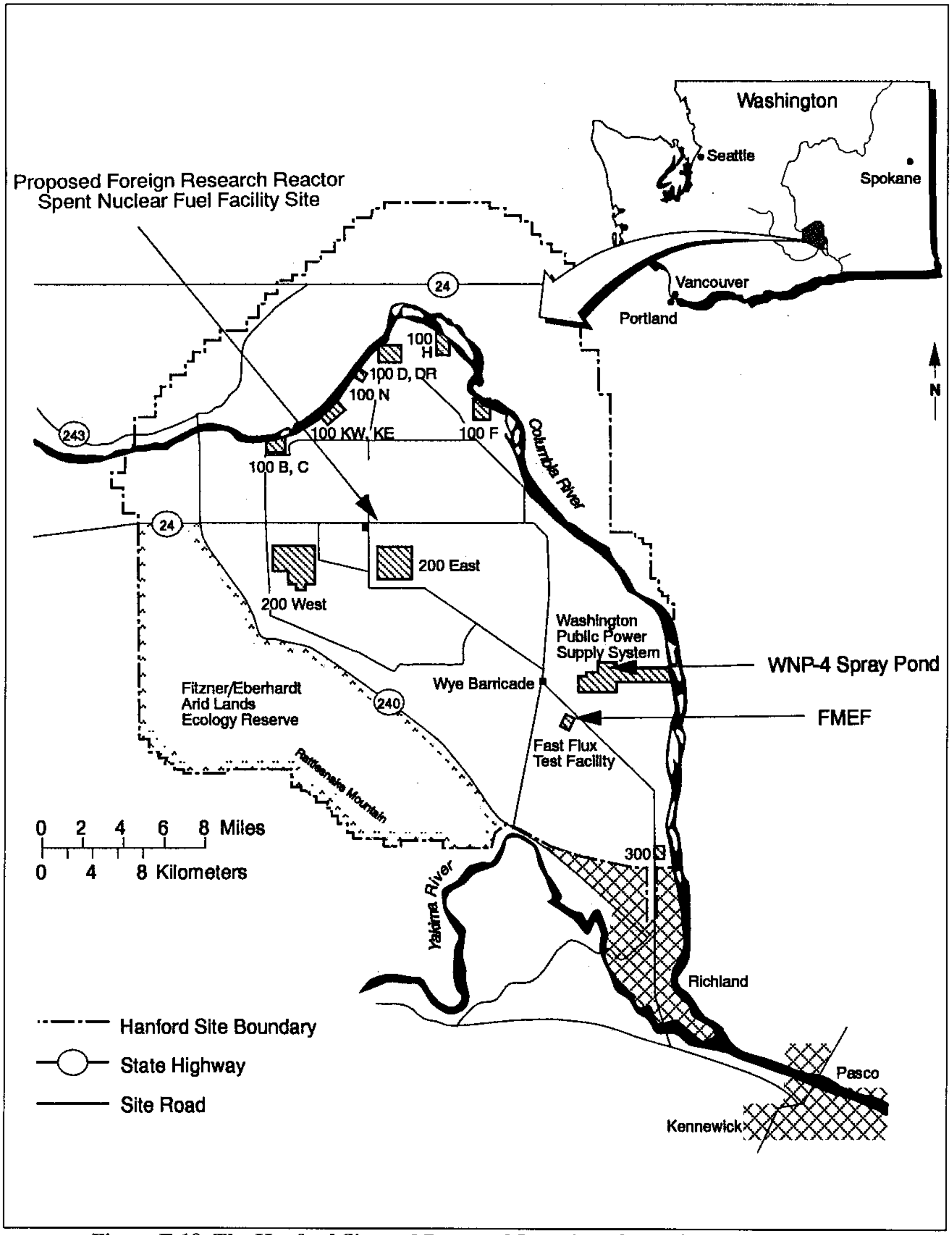


Figure F-19 The Hanford Site and Proposed Location of New Spent Nuclear Fuel Storage Facility

Table F-12 Description of Existing Spent Nuclear Fuel Facilities at Hanford Site

Facility	Description	Capacity for Foreign Research Reactor Spent Nuclear Fuel	Access
105-KE Basin[a]	Water storage pool; 38 m x 20 m x 6 m deep, concrete walls and floor, no sealant or liner	75% - 100% full	By rail, 27 metric tons crane, fairly restrictive
105-KW Basin[a]	Water storage pool, 38 m x 20 m x 6 m deep, concrete walls and floor, epoxy sealant, no liner	75% full; no space for foreign research reactor spent nuclear fuel	By rail, 27 metric tons crane, fairly restrictive
T Plant: Cell 4	Water storage pool, 4 m x 8.4 m x 5.8 m deep	50% full; no space for foreign research reactor spent nuclear fuel	By rail or truck All fuel handling remote
PUREX Plant: East end of 202A Bldg, plus Dissolver Cells A, B, and C[b]	Water storage pool, 9.5 m x 6.1 m x 5.2 m deep, Dissolver Cell sizes vary	No additional capacity	Shipment by rail 36 metric tons crane
Plutonium Finishing Plant: 2736-ZB Bldg.	Dry storage in 208 L[c]	No additional capacity	Shipment by truck
FFTF: Reactor in-vessel storage, interim decay storage, and fuel storage facility locations[b]	Liquid sodium pool storage (fuel storage facility is separate from reactor containment building, with limit of kilowatts/assembly)	More than 75% full; no space for foreign research reactor spent nuclear fuel	By truck 91 metric tons Crane
200 Area LL Burial Grounds: 218-W-4C Trenches 1 and 7; and 218-W-3A Trenches 8 and S6	Dry, retrievable storage, 13 lead-lined, concrete-filled 208 liter drums, soil covered, 22 concrete casks (1.66 m x 1.66 m x 1.22 m or 1.92 m high), soil covered, 39 EBR II casks (1.5 m high x 0.4 m diameter), soil covered; 1 Zircaloy Hull Container (152 cm long x 76 cm diameter)	Large additional capacity; not suitable for foreign research reactor spent nuclear fuel	By truck
308 Building Annex: Neutron Radiography Facility[b]	Built in late 1970s water storage pool, 2.8 m diameter x 6 m deep	Small additional capacity	Truck shipments 4.5 metric tons crane
324 Building: B and D Cells	Dry storage in air, B Cell: 6.7 m x 7.6 m x 9.3 m high (spent nuclear fuel uses 10% of floor space). D Cell: 4 m x 6.4 m x 5.2 m high (small part for fuel), thick concrete walls and floors with steel liners	Small additional capacity	Truck shipments only B Cell - 2.7 and 5.4 metric tons cranes; Airlock - 27 metric tons crane
325 Building: A and B Cells in 325A Radiochemical Facility; 325B Shielded Analytical Laboratory	Dry Storage in air 325A - 1.8 m x 2.1 m x 4.6 m high (typical cell) 325B - 1.7 m x 1.7 m floor area (typical cell)	Small additional capacity	Truck shipments only 325A - 27 metric tons crane 325B - 2.7 metric tons crane
327 Building: A-F and I Cells; Upper and Lower SERF; Dry Storage vault, EBR II cask, Large Basin	Dry storage in air, except for water in small basin; variety of cell sizes, but storage only for fuel research	Small additional capacity	No direct rail Truck shipments 13.5 and 18 metric tons cranes

FFTF = Fast Flux Test Facility; EBR = Experimental Breeder Reactor; PUREX = Plutonium Uranium Extraction

[a] *If 105-KE Basin fuel is consolidated with 105-KW Basin fuel, 105-KE Basin would be shut down. The storage capacity of 105-KW Basin would be increased by replacing all of the storage racks to allow multi-tiered stacking of fuel canisters and by making minor facility modifications.*

[b] *Facility is being shut down.*

[c] *One 55 gal drum.*

specifies that Hanford generated spent nuclear fuel will remain in storage at Hanford pending decisions on ultimate disposition. The second, is the Management of Spent Nuclear Fuel from the K Basins at the Hanford Site Draft EIS (DOE, 1995d) which was issued for comment in October 1995. This EIS addresses the location and method of managing Hanford spent nuclear fuel for up to 40 years or until decisions on ultimate disposition are made.

New facilities would be required for storage of foreign research reactor spent nuclear fuel. However, there may be some economies of scale achieved from overlap with the other spent nuclear fuel activities at the Hanford Site.

F.1.3.3.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Hanford Site for Foreign Research Reactor Spent Nuclear Fuel

The Hanford Site has concluded that there are no existing facilities available and ready for accepting foreign research reactor spent nuclear fuel (Bergsman et al., 1994). Consequently, the Hanford Site proposes the following strategies:

- construction of a "generic" modular dry vault or dry cask storage facility,
- construction of a "generic" wet storage and handling facility,
- modification and completion of the Fuel Maintenance and Examination Facility (FMEF) (located at the Fast Flux Test Facility) as a modular dry vault storage facility, and
- acquisition, modification, and completion of the Washington Nuclear Plant-4 Spray Cooling Pond (at the Washington Public Power Supply System) as a wet storage facility.

These facilities and their potential applications to foreign research reactor spent nuclear fuel storage are discussed in detail in Section F.3.

Figure F-20 displays the Hanford Site capacity for foreign research reactor spent nuclear fuel storage. The Hanford Site is not considered capable of immediately accepting foreign research reactor spent nuclear fuel because of the required construction of new facilities. The Hanford Site would have sufficient capacity for foreign research reactor spent nuclear fuel storage after new facility construction.

F.1.3.4 Oak Ridge Reservation

The Oak Ridge Reservation is located on approximately 140 km^2 (54 mi^2) of federally owned land near Knoxville, TN (DOE, 1995g). There are three primary plant complexes within the Oak Ridge Reservation:

- *Y-12 Plant:* produces various materials used for national defense purposes,
- *K-25 Site (formerly called the Oak Ridge Gaseous Diffusion Plant):* originally used for uranium enrichment and now an environmental management site, and
- *Oak Ridge National Laboratory (also known as X-10):* research and development into nuclear energy and other energy technologies.

The Oak Ridge National Laboratory has operated several small reactors for research and isotope production and, of the Oak Ridge Reservation sites, is the most familiar with spent nuclear fuel

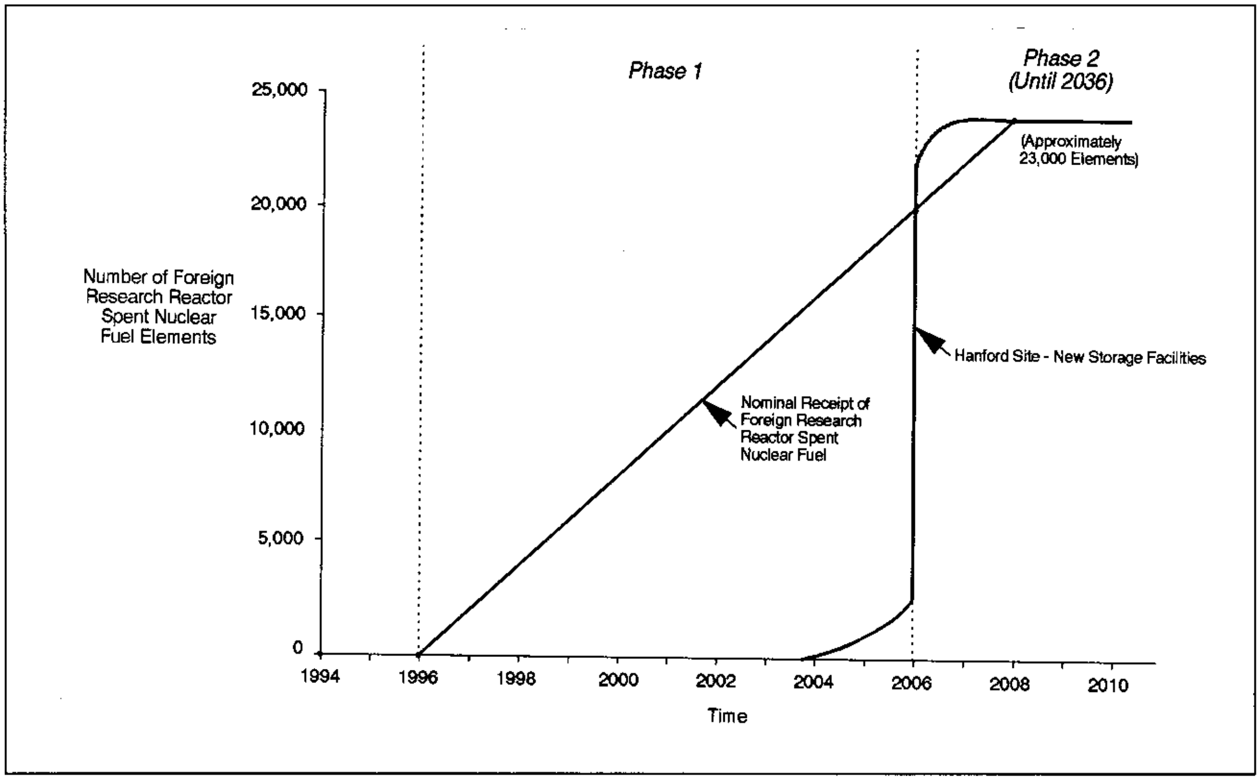


Figure F-20 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Hanford Site

requirements. A map of the Oak Ridge Reservation and its candidate sites for foreign research reactor spent nuclear fuel storage is presented in Figure F-21.

F.1.3.4.1 Spent Nuclear Fuel Activities at the Oak Ridge Reservation

Most Oak Ridge Reservation spent nuclear fuel activities occur at the Oak Ridge National Laboratory. The Oak Ridge National Laboratory has operated several small research reactors, all of which generate (or have generated) spent nuclear fuel. These reactors all have small fuel preparation and handling facilities associated with them ranging up to the single digit MTHM capacity. The spent nuclear fuel storage space is small, and most is either full or committed, with little excess capacity. The Oak Ridge National Laboratory also has hot cell and irradiated fuel examination facilities. Currently, only the High Flux Isotope Reactor is operating and generating spent nuclear fuel. More spent nuclear fuel facilities at Oak Ridge Reservation are presented in Table F-13.

F.1.3.4.2 Spent Nuclear Fuel Storage Facilities at the Oak Ridge Reservation

The Oak Ridge Reservation stores spent nuclear fuel in several small facilities. Most of these facilities are old and are unlikely to meet modern building code and seismic standards. The spent nuclear fuel facilities include the following structures:

- *Building 3525 - Irradiated Fuels Examination Laboratory:* This two-story brick structure was constructed in 1963. It houses hot cells and contains small quantities of irradiated research reactor fuel in the form of samples and targets.

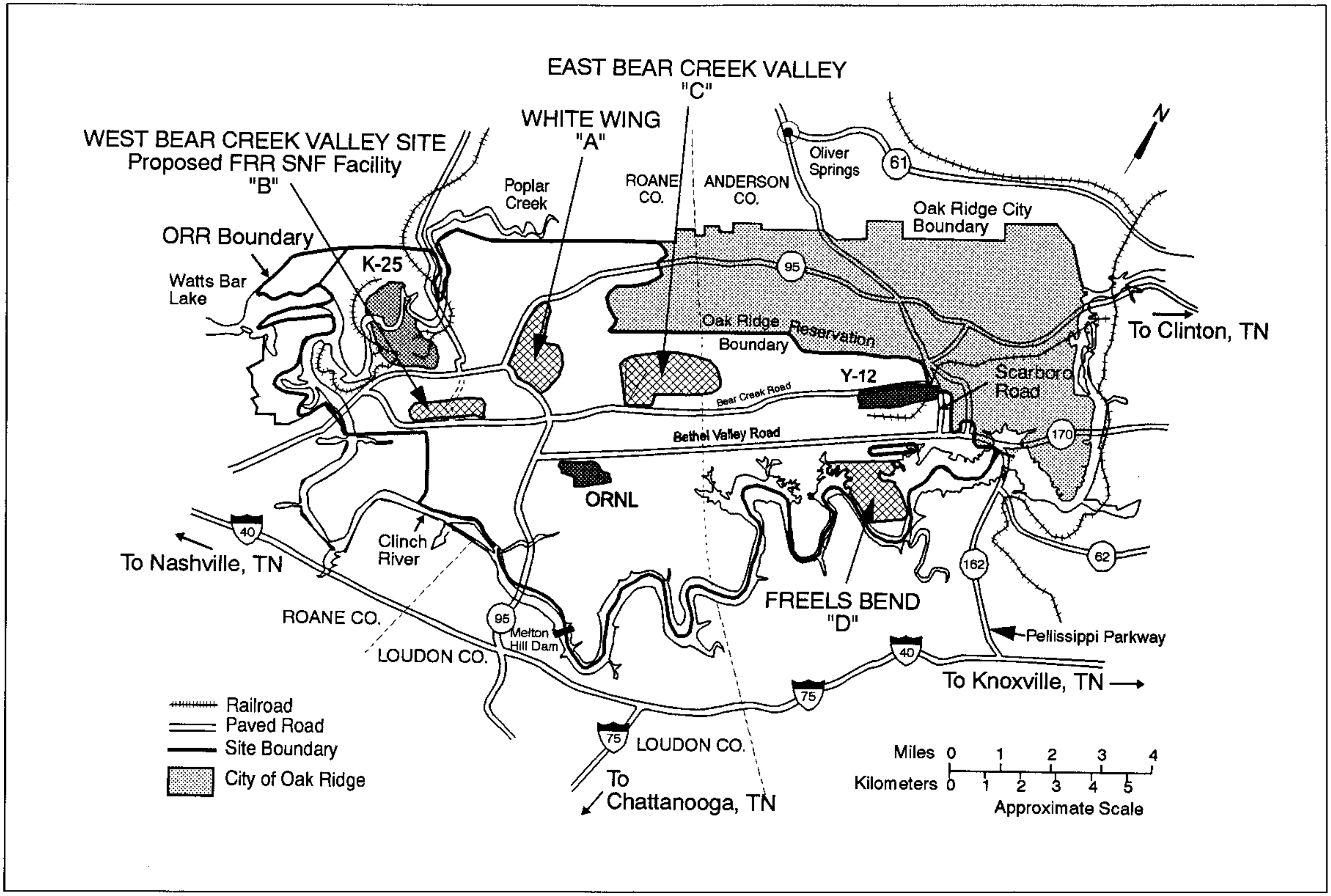


Figure F-21 Candidate Sites at the Oak Ridge Reservation for Foreign Research Reactor Spent Nuclear Fuel Storage

Table F-13 Major Spent Nuclear Fuel Facilities at the Oak Ridge Reservation

Facility	*Characteristics*	*Capacity for Foreign Research Reactor Spent Nuclear Fuel*	*Access*
Building 3525	Hot Cells	No, too small	Truck
Building 4501	Hot Cells	No, too small	Truck
Building 7827	Drywells	No space	Truck
Building 7920	Hot Cells	No, too small	Truck
Building 9720-5 (Y-12)	Warehouse	No, unirradiated fuel only	Truck
Other	Research Reactors	No, storage space near capacity	Truck

- *Building 4501 - High-Level Radiochemical Facility:* This facility dates from 1951 and contains hot cells for examining irradiated materials. This facility contains small quantities (several kg) of sectioned commercial fuel.
- *Building 7920 - Radiochemical Engineering Development Center:* This is a multi-purpose, hot cell facility for (relatively) large quantities of irradiated spent nuclear fuel. This facility supports target preparation and processing for the High Flux Isotope Reactor and contains samples and targets of research reactor spent nuclear fuel in dry storage.
- *Building 9720-5 (Y-12):* This is a large warehouse for storing and safeguarding unirradiated or low burnup HEU fuel. It currently contains around 0.2 MTHM.
- *Research Reactors:* There are five existing and one planned research reactor at the Oak Ridge Reservation. All of these reactors have small spent nuclear fuel storage basins nearby, and this capacity is essentially full. Only the High Flux Isotope Reactor is currently operating.
- *The Oak Ridge Reservation* also has several drywells such as Building 7827 and drum storage areas for irradiated fuel. Spent nuclear fuel would be relocated in accordance with actions of the DOE Programmatic SNF&INEL Final EIS (DOE, 1995g).

None of these locations have any significant capacity for the potential quantities of foreign research reactor spent nuclear fuel.

F.1.3.4.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Oak Ridge Reservation for Foreign Research Reactor Spent Nuclear Fuel

The Oak Ridge Reservation has plans for dry storage of spent nuclear fuel. This would be accomplished via a modular route at the High Flux Isotope Reactor location. This dry storage area could be extended almost indefinitely to accommodate the Oak Ridge Reservation's needs.

DOE is evaluating a spent nuclear fuel management complex for handling DOE spent nuclear fuel from other sites as an alternative in the DOE Programmatic SNF&INEL Final EIS (DOE, 1995g). The spent nuclear fuel management complex would include the following:

- Spent Nuclear Fuel Receiving and Canning Facility,
- Technology Development Facility,
- Interim Dry Storage Facility, and

- Expended Core Facility for Naval-type fuel similar to the one at Idaho National Engineering Laboratory.

The receiving and canning facility would receive spent nuclear fuel cask shipments from offsite and prepare the spent nuclear fuel for dry storage. The facility incorporates a pool (wet) storage facility for cooling spent nuclear fuel (tentatively identified as a 5-year period) prior to placement into dry storage, as necessary. The technology development facility would investigate the applicability of dry storage technologies and pilot scale technology development for disposal for various types of spent nuclear fuel. The interim dry storage area would consist of passive storage modules to safely store the spent nuclear fuel for 40 years. Naval fuel would be examined at the Expended Core Facility prior to interim storage. The total land required for the facility, including a buffer zone, is approximately 36 ha (90 acres).

The proposed site for the spent nuclear fuel facilities is located in the West Bear Creek Valley Area, in the western portion of the Oak Ridge Reservation site. This area of the Oak Ridge Reservation is currently in the Natural Areas land use category and is designated for future Waste Management land use. Land uses bordering on the Oak Ridge Reservation in this area are primarily agricultural farmland and commercial forest, with sparsely located residences (i.e., low population density).

Environmental, safety, and health consequences are calculated to be negligible from the spent nuclear fuel facilities, although a preliminary design and/or layout is not provided. Releases of krypton-85, chlorine, and hydrogen fluoride are included in the analysis for incident-free operations, but the source of these emissions is not reported. Facility budgetary requirements are not delineated.

Foreign research reactor spent nuclear fuel represents less than one percent of the DOE spent nuclear fuel quantities in terms of mass and, thus, its effect would be minimal as compared to the other fuels. The foreign research reactor spent nuclear fuel contribution to the operational consequences and its costs are not delineated. Figure F-22 summarizes foreign research reactor spent nuclear fuel capacity at the Oak Ridge Reservation. New facility construction would be required for foreign research reactor spent nuclear storage.

F.1.3.5 Nevada Test Site

The Nevada Test Site is located in the southeastern part of the State of Nevada, and is used as the on-continent site for nuclear weapons testing (DOE, 1995g). The Nevada Test Site encompasses approximately 3,500 km^2 (1,350 mi^2) of desert land, with flats, mesas, and mountain ridges (Figure F-23). Essentially no permanent surface waters exist, and the depth to groundwater routinely exceeds 330 m (1,000 ft). The Nellis Air Force Base Range surrounds the Nevada Test Site to the north, east, and west; and, with the Tonapah Test Range, provides a 24 to 104 km (15 to 65 mi) buffer zone between the Nevada Test Site and public lands. The Bureau of Land Management owns land on the southern and southwestern borders of the Nevada Test Site. Principal access to the site is via the town of Mercury, on the southeastern corner. Las Vegas is approximately 104 km (65 mi) from this corner of the Site.

Activities at the site have included nuclear weapons testing, nuclear reactor tests, nuclear rocket engine development, and waste management. Current activities include nuclear weapons-related activities (e.g., emergency search teams, arms control/verification, etc.), low-level waste/low-level mixed waste disposal, and site characterization for commercial spent nuclear fuel disposal.

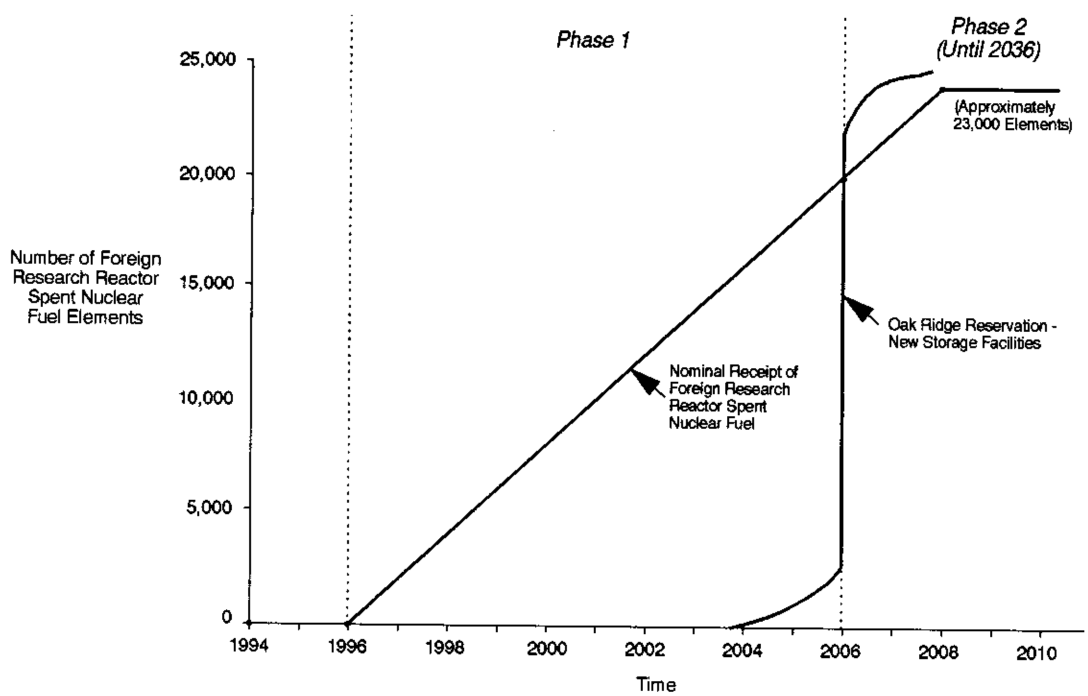


Figure F-22 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Oak Ridge Reservation

F.1.3.5.1 Spent Nuclear Fuel Activities at the Nevada Test Site

The Nevada Test Site has several existing facilities that could be useful for spent nuclear fuel management. These facilities were principally used for nuclear rocket engine development and are located at Jackass Flats, in a southern portion of the Nevada Test Site called the Nevada Research & Development Area (Cosimi, 1994; Chandler et al., 1992; Gertz, 1994; Hynes, 1994; Reed, 1994). The facilities include several large hot cell and fuel examination "shops," with large cranes and manipulators. At least two of these facilities appear to be ideally suited for handling and storing foreign research reactor spent nuclear fuel after relatively minor upgrades and refurbishments. Table F-14 summarizes the capabilities of these facilities for foreign research reactor spent nuclear fuel.

The Engine Maintenance and Disassembly (E-MAD) facility was originally constructed for the assembly and preparation of nuclear rocket engines for testing, refurbishment of activated engines, and disassembly and inspection of tested engines and components. The facility is designed for remote handling and examination of highly radioactive components. The building is a T-shaped, multi-storied structure, with overall dimensions of 85 x 107 m (280 ft x 350 ft) (Figure F-24). Numerous hot cells exist, with remote handling and transfer equipment, and the largest hot cell is 20 m wide x 45 m long x 23.5 m high (66 ft wide by 146 ft long and 77 ft high). Typically, 1.5 m (5 ft) thick concrete walls provide the shielding. Material transfer capabilities include several 36 metric tons (40 ton) cranes and a cask handling system of

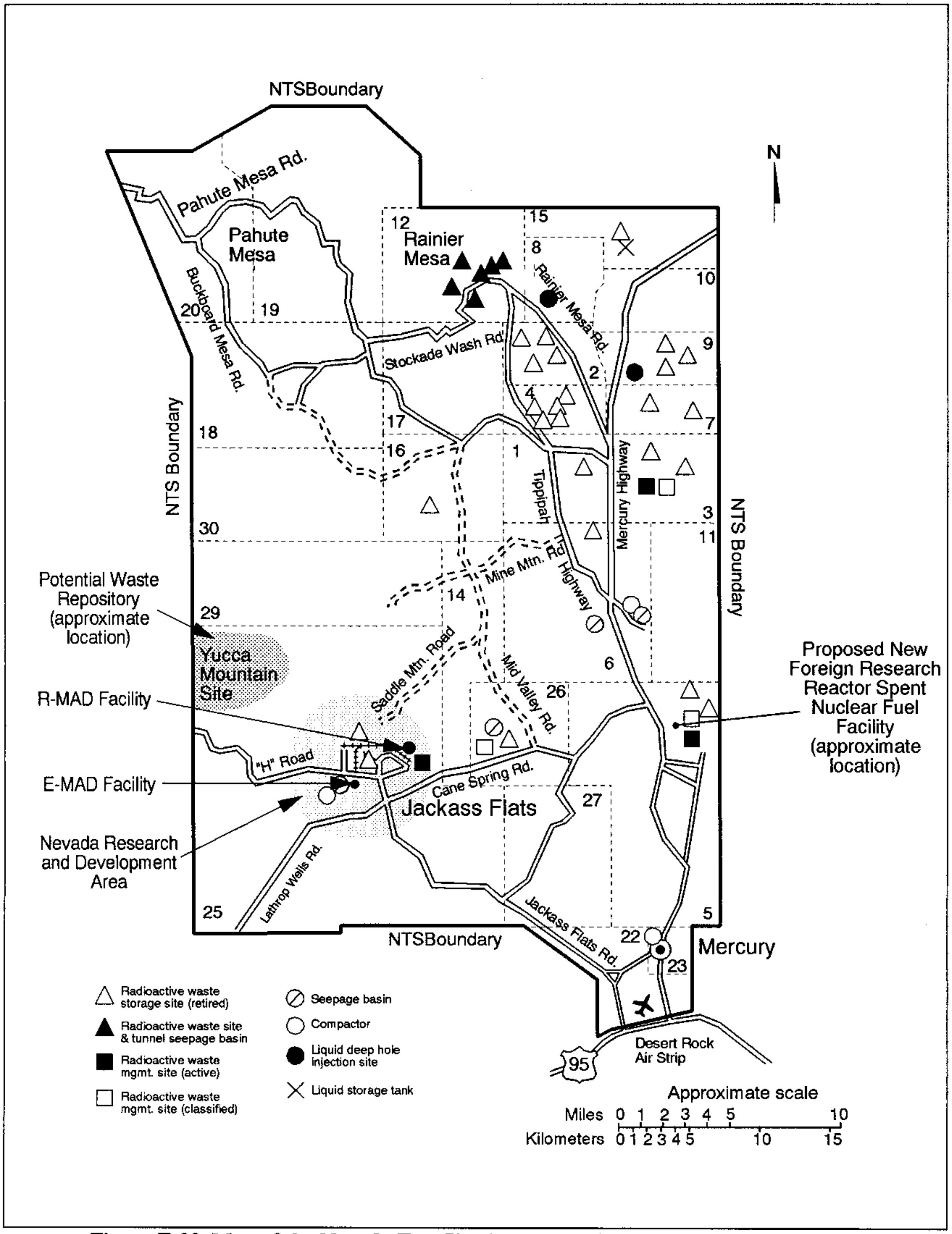


Figure F-23 Map of the Nevada Test Site Area 5 and Area 25 as Potential Foreign Research Reactor Spent Nuclear Fuel Storage Areas

Table F-14 Major Spent Nuclear Fuel-Capable Facilities at the Nevada Test Site

Facility	Characteristics	Capacity for Foreign Research Reactor Spent Nuclear Fuel	Access
E-MAD	Large hot cell, with smaller hot cells. Main hot cell area is $895m^2$ ($9,600\ ft^2$)	Yes, as either a vault or a staging facility for dry casks, after 1-2 years refurbishments, 25,000 elements	Truck
R-MAD	Large hot cell, with smaller hot cells. Main hot cell area is $223m^2$ ($2,400\ ft^2$)	Yes, as a staging facility for dry casks or a small vault, after 1-3 years refurbishment, 25,000 elements	Truck

approximately 91 metric tons (100 tons) capacity. The heating, ventilation, and air conditioning systems for the hot cell areas maintain negative pressure and exhaust through High Efficiency Particulate Air filters.

The E-MAD facility is currently unused and last saw service during the 1980s for commercial spent nuclear fuel storage experiments (e.g., Climax Mine Project) (Gertz, 1994). Thirteen commercial spent nuclear fuel assemblies were tested in casks and drywells. The E-MAD facility was subsequently used to load transportation casks for shipment of the spent nuclear fuel to Idaho. Several of these spent nuclear fuel storage casks remain at the site (Hynes, 1994). The Los Alamos National Laboratory assessment (Chandler et al., 1992) considers the facility to require only minor upgrades and routine maintenance.

The Reactor Maintenance and Disassembly facility is located a short distance from the E-MAD facility. This facility contains two (contact) assembly bays and one remotely operated hot disassembly bay. The hot bay dimensions are 18 x 12 x 18 m (60 by 40 by 60 ft) high, with 1.8 m (6 ft) thick walls for shielding. A transfer system connects six hot cells to the hot disassembly bay. The Los Alamos National Laboratory assessment (Chandler et al., 1992) found the Reactor Maintenance and Disassembly facility to require a minor upgrade.

F.1.3.5.2 Spent Nuclear Fuel Storage Facilities at the Nevada Test Site

At the present time, the Nevada Test Site is not storing spent nuclear fuel. As noted, facilities in the Jackass Flats area have handled spent nuclear fuel in the past and could be adapted to accommodate foreign research reactor spent nuclear fuel and serve as the nucleus of a spent nuclear fuel storage facility. The E-MAD and Reactor Maintenance and Disassembly facilities appear to have sufficient size and design for accommodating all of the foreign research reactor spent nuclear fuel in a dry storage mode, either vault or cask, and for accomplishing any required transfer, examination, and canning operations.

F.1.3.5.3 Planned or Potential Spent Nuclear Fuel Storage Facilities at the Nevada Test Site for Foreign Research Reactor Spent Nuclear Fuel

Besides the Area-25 facilities, DOE evaluated an elaborate spent nuclear fuel handling system as an alternative in the DOE Programmatic SNF&INEL Final EIS (DOE, 1995g). The spent nuclear fuel management complex would be located in Test Area 5, near the eastern border of the site, and in the general proximity of the low-level waste/low-level mixed waste disposal areas. The spent nuclear fuel complex would include:

- Spent Nuclear Fuel Receiving and Canning Facility,
- Technology Development Facility,
- Interim Dry Storage Area, and
- Expended Core Facility, similar to the one at Idaho National Engineering Laboratory.

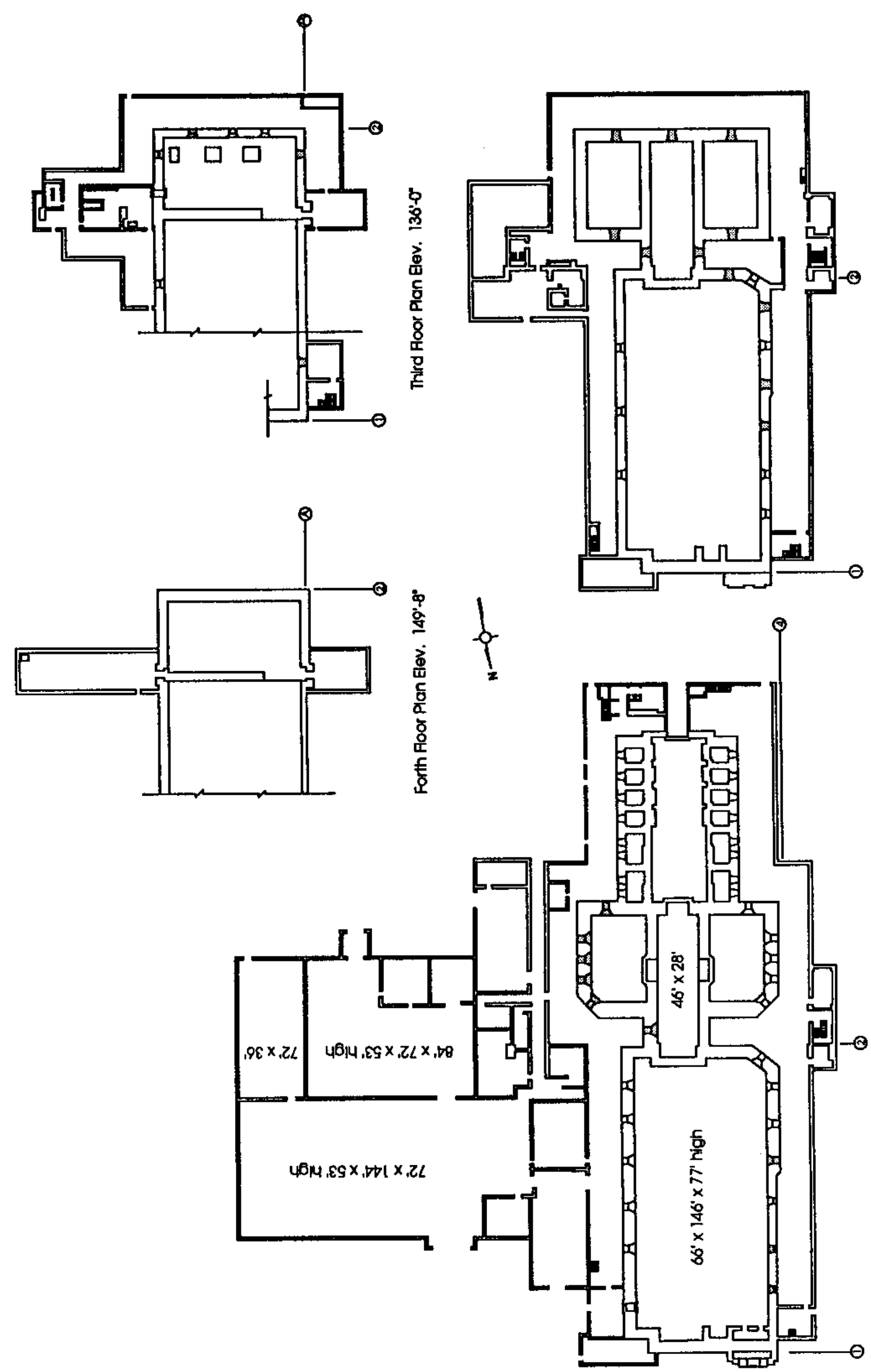


Figure F-24 Schematic of the Engine Maintenance and Disassembly Facility at the Nevada Test Site

The receiving and canning facility would receive spent nuclear fuel cask shipments from offsite and prepare the spent nuclear fuel for dry storage. The facility incorporates a pool (wet) storage facility for cooling spent nuclear fuel (tentatively identified as a 5-year period) prior to placement into dry storage, as necessary. The technology development facility would investigate the applicability of dry storage technologies and pilot scale technology development for disposal for various types of spent nuclear fuel. The interim dry storage area would consist of passive storage modules to safely store the spent nuclear fuel for 40 years. Naval fuel would be examined at the Expended Core Facility prior to interim storage. The total land required for the facility, including a buffer zone, is approximately 36 ha (90 acres).

Environmental, safety, and health consequences are calculated to be negligible from the spent nuclear fuel facilities, although a preliminary design and/or layout is not provided. Releases of krypton-85, chlorine, and hydrogen fluoride are included in the analysis for incident-free operations, but the source of these emissions is not reported. Facility budgetary requirements are not reported.

Foreign research reactor spent nuclear fuel represents less than 1 percent of the DOE spent nuclear fuel quantities in terms of mass (i.e., potential source term), and about 10 percent in terms of volume. Thus, its effect would be minimal as compared to the other fuels. The foreign research reactor spent nuclear fuel contribution to the operational consequences and its costs are not delineated.

Figure F-25 summarizes the Nevada Test Site storage capabilities for foreign research reactor spent nuclear fuel. The Area-25 facilities could all receive and provide for dry storage of foreign research reactor spent nuclear fuel close to the proposed Yucca Mountain repository. It should be noted that these facilities have comparable shielded floor areas and volumes as compared to the generic modular dry vault

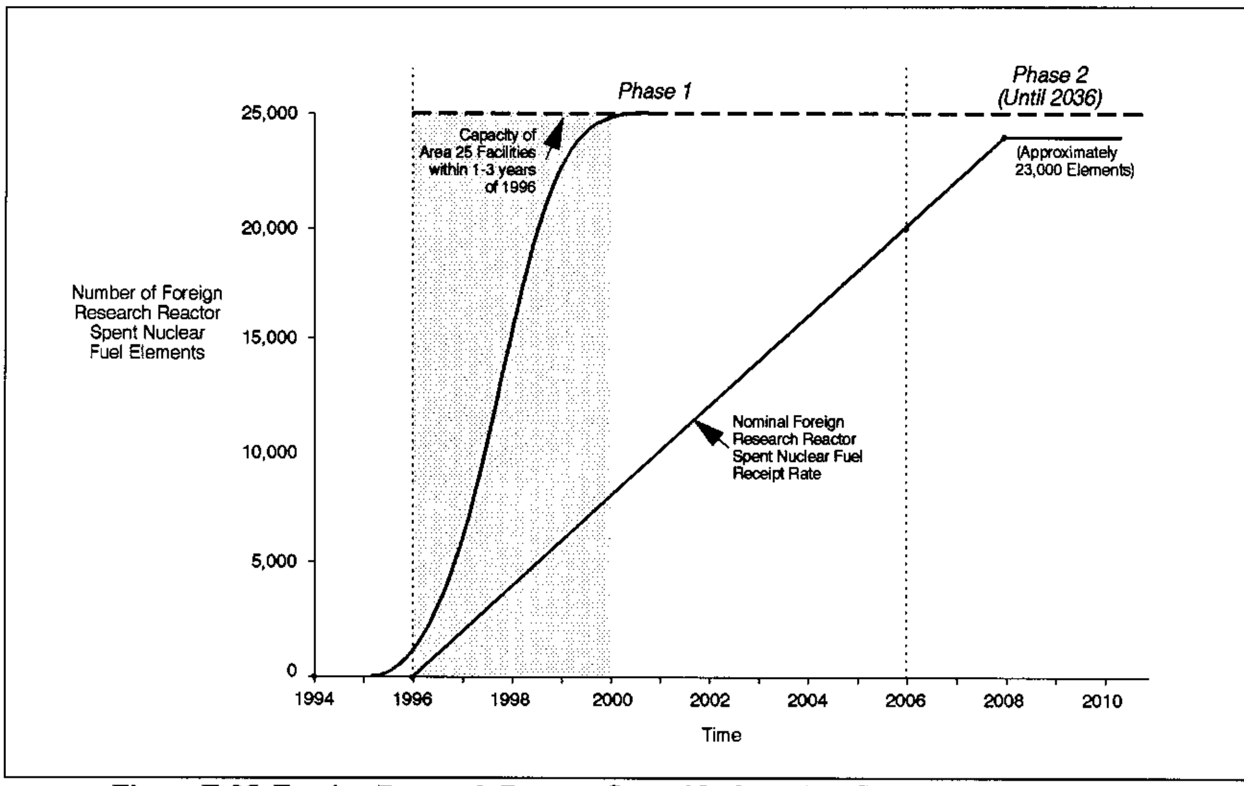


Figure F-25 Foreign Research Reactor Spent Nuclear Fuel Storage Capacity at the Nevada Test Site

for foreign research reactor spent nuclear fuel discussed in Section F.3. Alternatively, new facilities could be built, but these would require a longer transportation path to the proposed Yucca Mountain repository.

F.1.3.6 Storage at Overseas Facilities

Currently, foreign research reactor spent nuclear fuel is being stored in wet pools at foreign research reactor sites. These pools are approaching the levels of their capacity, which is why the foreign research reactor operators would like the United States to accept their spent nuclear fuel. An alternative being considered by DOE is foreign research reactor spent nuclear fuel storage at overseas facilities. Several facilities exist in Europe for contractual storage of both commercial and research reactor spent nuclear fuel for a fee, including:

- British facilities at Dounreay, Scotland and Sellafield, England. The former has several small pools for research reactor fuels, while the latter has several large pools with a capacity of 3,000 MTHM for commercial spent nuclear fuel (Bonser, 1994).
- French facilities at La Hague, with several large pools having a total capacity of 14,000 MTHM for commercial spent nuclear fuel (Nuclear Fuel, 1993); facilities at Marcoule, for research and metallic spent nuclear fuel.

Electricite De France has also announced its intention of constructing a commercial spent nuclear fuel wet storage facility with a capacity of 12,000 MTHM (Nuclear Fuel, 1994b). Dry storage of spent nuclear fuel is also being considered.

These facilities are predominantly stainless-steel lined wet storage pools that meet modern seismicity and confinement standards and maintain good water chemistry. Wet storage pools designed for commercial spent nuclear fuel could, after license modification and new rack installation, store foreign research reactor spent nuclear fuel. These overseas wet storage pools are similar in design and layout to the generic wet storage facility discussed in Section F.3.

F.1.4 Vitrified Waste Storage Facilities

If foreign research reactor spent nuclear fuel is processed, the resulting high-level waste would be vitrified and placed into stainless steel canisters. The Savannah River Site is the only domestic site that currently has a storage facility designed and built for storing vitrified high-level waste from the processing of spent nuclear fuel. This facility is termed the Glass Waste Storage Building, and it is located immediately adjacent to the Savannah River Site vitrification facility (the Defense Waste Processing Facility), in the S-Area of the site near the H-Area processing facilities (DOE, 1994g). Figure F-26 provides a general overview of the facilities in the S-Area. Figure F-27 displays a general layout of the building. The Glass Waste Storage Building is designed to accommodate the standard Defense Waste Processing Facility vitrified waste canister (Figure F-28). The existing building has space for 2,286 of these canisters. A second, almost identical building, is planned for construction starting in 2007. Additional buildings may be built, up to a total interim storage capacity of 10,000 canisters if delays in the Federal Repository Program are encountered (DOE, 1994g). The Defense Waste Processing Facility/Glass Waste Storage Building area does not currently include a cask receiving/shipping facility, but one is planned for future construction.

The facility is relatively simple in design and operation. It consists of a structure enclosing a concrete floor that functions as the charging face to the vault beneath it. Shield plugs are removed from the floor to provide access to storage tubes in the vaults that would contain the canisters. Each storage tube contains

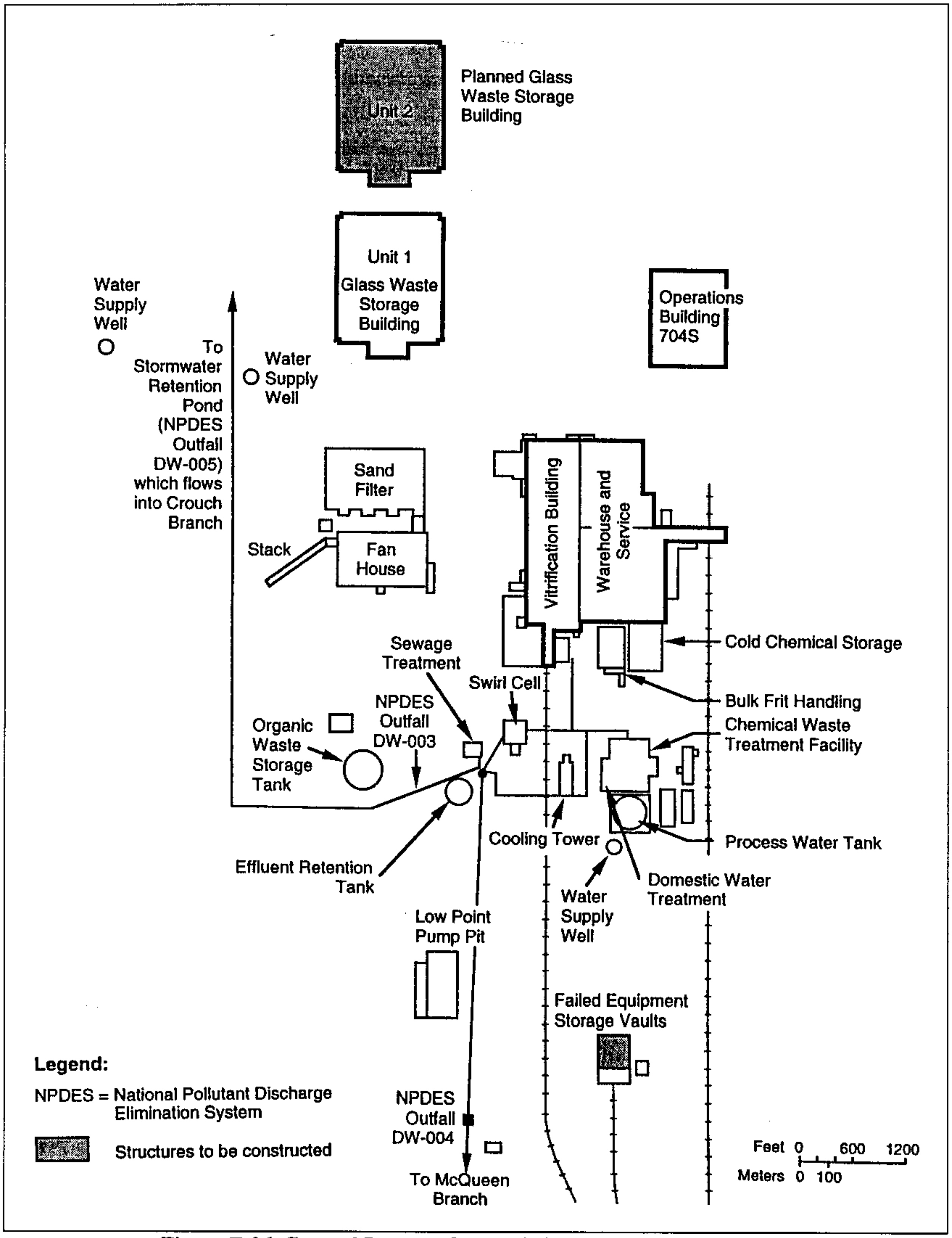


Figure F-26 General Layout of the Existing Vitrification Facilities at the Savannah River Site

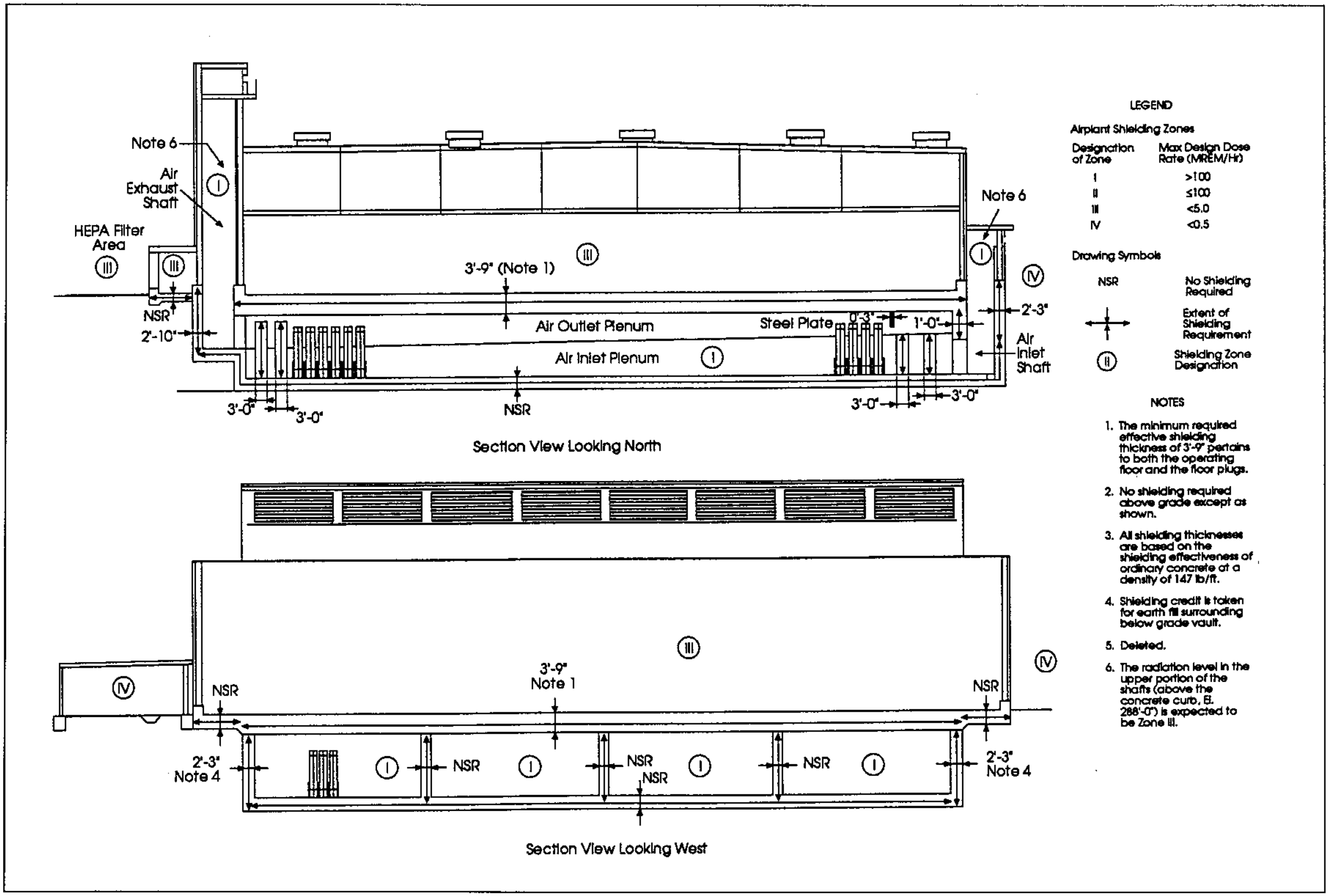
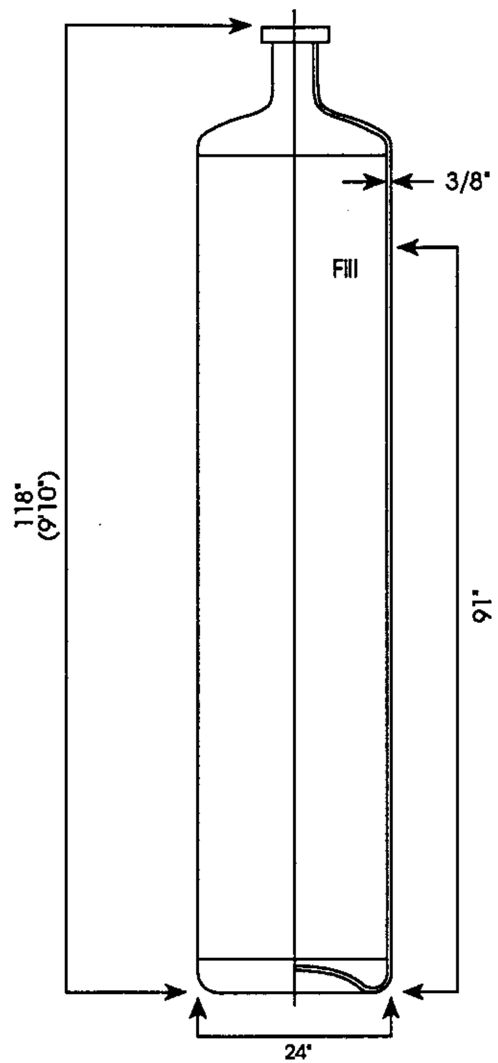


Figure F-27 Layout of the Glass Waste Storage Building



Material:	304L Stainless Steel
Empty Weight:	1,100 lb
Net Weight:	3,540 lb (Glass)
Weight of Radionuclide:	41 kg (90 lb)
Activity:	8.3×10^{15} Bq (223,000 Ci)
Decay Heat:	660 W
Radiation Field (at Surface):	6,300 rad/hr
Surface Contamination:	Less than 10^{-4} μCi/cm^2

Figure F-28 Typical Defense Waste Processing Facility Glass Waste Canister

two canisters, stacked vertically. The vault area consists primarily of steel-reinforced concrete and is designed to resist all earthquake and severe weather incidents.

Radioactive decay heat from the canisters is removed by the Glass Waste Storage Building's forced air fan exhaust system. The exhaust air is drawn around the canisters and then exhausted through the building's High Efficiency Particulate Air filtered ventilation system and discharged to the atmosphere via a stack. No condensate is expected to form, although the building does include a sump for exhaust air condensate. No radioactivity is expected in the exhaust air or in any condensates that might form.

During operation, a special dedicated transporter vehicle moves the canister from the Defense Waste Processing Facility vitrification building to the Glass Waste Storage Building in a shielded transporter. The transporter's cask is placed over the appropriate vault borehole, the shield plug is removed, and the canister is lowered via a crane mechanism into the borehole. The shield plug is replaced, and the transporter returns to the vitrification plant for the next shipment.

Several overseas facilities also exist for vitrified waste storage at Marcoule (France), La Hague (France), and Sellafield (England) (COGEMA, 1994a and 1994b; BNFL, 1994a and 1994b). These facilities are designed as natural circulation vaults and, thus, do not require fans for storage cooling. These vaults use a smaller canister, with several thousand currently in storage.

F.2 Storage Technology Evaluation Methodology

The selection of a spent nuclear fuel storage technology for foreign research reactor spent nuclear fuel requires a multi-disciplinary approach including the evaluation of, at the minimum, the environmental impact of alternatives and the following key design and performance areas:

- chemical compatibility,
- subcriticality assurance,
- shielding effectiveness,
- structural integrity (i.e., containment),
- thermal performance,
- ease of use,
- cost, and
- regulatory basis and licensing.

Other factors that may affect the decision process are whether the design has been previously licensed and actually used to store spent nuclear fuel, and its perceived ability to meet applicable regulations and standards if it has not yet been licensed.

Two principal types of spent nuclear fuel storage can be used for foreign research reactor spent nuclear fuel, wet and dry. Wet storage denotes the immersion of fuel in a pool of water, which performs the dual functions of shielding/leaking radionuclide removal and decay heat removal, but which relies on active systems. Dry storage encompasses a wide spectrum of structures that house the fuel in a dry inert gas environment, with an emphasis on passive system design and operation.